Atmospheric dispersion and environmental consequences. Exposure from radioactive plume pathways

Jensen, Per Hedemann

Publication date:
1992

Document Version
Publisher's PDF, also known as Version of record

Risø-M-2849(EN)

Atmospheric Dispersion and Environmental Consequences

Exposure from Radioactive Plume Pathways

Per Hedemann Jensen

Risø National Laboratory, DK-4000 Roskilde, Denmark
November 1992

Risø-M-2839(EN)

Atmosphedic Dispersion and Environmental Consequences

Exposure from Radioactive Plume Pathways

Per Hedemann Jensen

Risø National Laboratory, Roskilde, Denmark
November 1992

Abstract Methods are described for assessing early radiation doses due to atmospheric releases of radionuclides, i.e. inhalation and external exposure from the plume and from deposited activity. Data to be used in these assessments are presented. The purpose of the present work is to evaluate methods and data that could be used in emergency situations as well as for emergency planning purposes.

The most important direct pathways following a release of airborne radionuclides to the atmosphere are the inhalation pathway and the external exposure pathway from ground-deposited activity. For long-lived radionuclides like ^{134}Cs and ^{137}Cs the committed effective external dose from deposited activity is 1-2 orders of magnitude larger than the committed effective dose from inhalation. Similarly, the committed effective dose from inhalation is 1-2 orders of magnitude larger than the external γ-dose originating directly from the plume.

This work has been performed as a part of the Nordic Safety Research Programme (NKS) under the Section "Emergency in abnormal radiation situations" (BER). The purpose of the BER-programme is to evaluate systematically those parts of emergency preparedness that need to be harmonized within the Nordic countries as a basis for uniform action in emergency situations.

The present report of the NKS-programme, BER-12(92)1, has been prepared within Risø-project No. 02696-00.

ISBN 87-550-1613-8
ISSN 0418-6453

Print: Pitney Bowes Management Services Danmark A/S, 2008

Contents

1 Introduction

In the event of a nuclear accident which results in the release and dispersion of radioactive materials into the environment, the effective implementation of measures for protecting the public will largely depend on the adequacy of **advance** preparation, including the preparation of an **emergency response plan** to control and limit the consequences of the accident. The accident consequences will be specific to each type of accident, and there is unlikely to be a single accident sequence which can be identified as the only one upon which to base emergency response plans. The accident scenarios for planning purposes must cover a wide range, even though throughout this range accidents may have an extremely low probability of occurrence. These scenarios include those where the off-site consequences are expected to be small, to those having significant consequences off the site.

The emergency response plan should be based on a risk assessment of the nuclear facility in question. Consideration should be given in particular to accidents leading to radiation hazards to man and the environment via airborne and ground-deposited radionuclides. Accident analysis embraces both the determination of the size, type, nature and point of release of radioactive material, and a projection of the consequences to off-site persons and the environment in the absence of any response. Accident analyses can be specific to an individual facility and can also be generic for any given type of nuclear facility.

For atmospheric releases from nuclear power facilities, three dominant contributors to short time exposures have been identified: (1) whole body exposure from external γ-radiation; (2) thyroid exposure from inhalation of or ingestion of radioiodines; and (3) exposure of other organs (e.g. lungs) from inhalation of radioactive materials. Any of these exposure modes could dominate the total effective doses, depending upon the type of radionuclides in the release and their chemical and physical forms.

The radionuclides in nuclear reactors exist in an variety of physical and chemical forms of varied volatility. The characteristics of these materials show quite clearly that the potential for releases to the environment decreases significantly in the order: (1) gaseous materials; (2) volatile solids; and (3) non-volatile solids. Because of the importance of gaseous releases there is an emphasis on the development of plans relating to the release of noble gases and of volatiles such as iodine. However, particulate materials should also be considered.

During the last decade there have been a number of research projects on atmospheric dispersion and environmental consequences. The most important has been performed within the CEC Radiation Protection Programme, especially the so-called MARIA-programme (**M**ethods for **A**ssessing the **R**adiological **I**mpact of **A**ccidents) in the years 1983-1989 [1]. Major research has also been conducted in the Nordic countries within the AKTU-programme funded by the Nordic Council of Ministers, via the Nordic Liaison Committee on Atomic Energy (NKA) in the years 1985-1989 [2].

Both programmes have contributed considerably to the continued development of methods and models and have improved the assumptions and model parameters for nuclear accident consequence assessment. New information has also emerged after the Chernobyl accident, both in the affected republics in the former USSR and in Western Europe.

From a Nordic **emergency planning** point of view, the purpose of the present project is to collect and evaluate data for use in assessing radiation doses directly from the plume and ground deposition exposure pathways in the light of both

the AKTU- and MARIA-programmes and the Chernobyl experience. The indirect exposure pathways such as ingestion of contaminated foodstuffs and inhalation of resuspended materials are not included in this project but are dealt with in the project BER-121 [34].

2 Historical background

2.1 Safety studies in the past

The first official report of importance in which an assessment of a severe nuclear power plant accident was studied was published by the U.S. Atomic Energy Commission in 1957 in *WASH-740* [3]. The report was prepared in order to assess liabilities in connection with nuclear accidents. A number of crude assumptions were made, and the report focused on the consequences of severe accidents. The accident probabilities were not estimated and, consequently, the risk dimension of the accidents was neglected.

In the following years several studies on reactor accident consequences were made. In the Nordic countries the *Urban Siting Study* [4] from Sweden was published in 1974, one year before the publication in 1975 of the *U.S. Reactor Safety Study* [5] in WASH-1400 (the Rasmussen Report) by the U.S. Nuclear Regulatory Commission. Later studies were all more or less influenced by WASH-1400, including the *German Reactor Safety Study* [6] and studies in the Nordic Countries such as the Norwegian *Nuclear Power and Safety* [7], the Finnish *Risk-Benefit Evaluation of Nuclear Power Plant Siting* [8] and different Danish/Swedish studies on the potential consequences of severe accidents at the Swedish nuclear power plant Barsebäck [9,10,11].

From the mid-seventies until the early eighties an informal Nordic working group, SNODAS (**S**ammenligning af **No**rdiske modeller til beregning af **D**oser og **A**tmosfærisk **S**predning (Coordination of Nordic models for calculation of doses and atmospheric dispersion)) made a valuable contribution to the harmonization of existing and newly developed dispersion/dose models in the Nordic countries [12,13,14,15,37], and the work resulted in an important improvement of Nordic dose models. Subsequently, the OECD/NEA adopted this work form for its international model comparison study [16].

The main conclusions of the above-mentioned reactor safety studies are:

(a) The possible consequences of potential nuclear accidents are predicted to be *no larger*, and in many cases much smaller, than those of non-nuclear accidents. The consequences are predicted to be smaller than those found in previous studies which deliberately maximized estimates of these consequences.

(b) The likelihood of reactor accidents is *much smaller* than that of many non-nuclear accidents having similar consequences. Non-nuclear accidents such as fires, explosions, toxic chemical releases, dam failures, hurricanes and tornadoes, are all much more likely to occur and can have overall consequences comparable to, or larger than, those of nuclear accidents.

(c) For elevated releases, a *heavy* local ground contamination density could occur at *large distances* from the release point due to wet deposition by rain. For example, several hours of brisk winds with stability category A followed by a near calm or rain can result in high ground dose rates, and early fatalities might occur beyond 25 kilometers from the release point under such circumstances.

These "hot spots" will, however, occur with a low probability because, at a specific distance, rain has a relatively low probability of occurrence.

(d) For *low probability* nuclear accidents, e.g. those with a *large release* to the atmosphere, the average ratio of early to late cancer fatalities would be on the range of 0.05 - 0.01 for a probability range for these consequences on the order of 10^{-9} - 10^{-7} per reactor year. For probabilities higher than this, i.e. where the release is lower, the ratio of early to late fatalities would decrease, and above a given probability no early fatalities would occur at all.

The above conclusions have been supported by both the TMI (**T**hree **M**ile **I**sland) and Chernobyl accidents. Both were core-melt accidents of the low probability type. The health consequences for the surrounding populations are expected to be low, for the TMI accident they were negligible. The late consequences of the Chernobyl accident in the affected areas in the Republics of Byelorussia, Ukraine and Russia in a population of around 800,000 people are expected to be of the order of 50 late cancer cases per year averaged over a 50-year period [17]. This corresponds to an average individual risk of less than 10^{-4} y^{-1}. If the Chernobyl reactors had been built with a containment, the health consequences might have been considerably lower, even as low as for the TMI accident.

2.2 CEC and Nordic co-operation

Nordic co-operation in the area of Accident Consequence Assessment takes place within the Nordic Safety Programme. This work was partially funded by the Nordic Council of Ministers, via the Nordic Liaison Committee on Atomic Energy (NKA/NKS). In the years 1985-1989 this programme included as a main project studies on releases of activity from severe accidents, its dispersion and environmental impact. The project was subdivided in two separate parts named **AKTI** (**AKT** for active material and **I** for inside the reactor containment) and **AKTU** (**U** for outside the reactor containment).

The AKTU-programme addressed the methods and models for calculating environmental dispersion of radioactive materials after a release and the radiological consequences to man as well as different mitigating actions and their associated socio-economic consequences.

The AKTU-programme has contributed to the continued development of methods and improvement of assumptions and parameters used within the area of nuclear accident consequence assessment, so that these consequences can be assessed with a higher degree of certainty. The programme has also contributed to improved understanding of the problems involved in more general terms. Of particular importance has been the investigation of problems where very little work has been done previously, such as winter conditions, uptake in animals grazing in mountain areas and various types of mitigating actions [2].

The CEC Radiation Protection Research Programme [1] during 1985-1989 was divided into six scientific sectors, and the MARIA-project was a part of Sector B: *Behaviour and Control of Radionuclides in the Environment.* The MARIA-project, launched in 1983, was continued and the consequence assessment code, COSYMA (**C**ode **Sy**stem of the **Ma**ria Project) has been developed. Some post-Chernobyl actions (long-distance atmospheric transfer and accident scenarios) complemented the project and this led to a more comprehensive consideration of complex terrain configurations and the impact of large topographical formations. Furthermore, a computer-based expert system (decision-aiding modules) was developed for assessing the positive and negative consequences of introducing countermeasures.

As a result of the programme, more accurate input data is now available to be used for developing predictive models and for making safety evaluations for nuclear plants and for waste disposal. A better basis has also been provided to evaluate the effects of accidents and as background for decisions in accident situations.

In the present CEC-programme on radiation protection (1991-1994), there are two CEC/Russian Joint Study Projects (JSPs) and five Experimental Collaboration Projects (ECPs) all of which are relevant for emergency planning purposes. These seven projects form part of the CHECIR-programme (**Che**rnobyl **C**enter for **I**nternational **R**esearch).

3 Exposure pathways from radioactive plumes

3.1 External radiation

In a continuous release of airborne materials, the material disperses downwind as a plume. The concentration at ground level at specific distances from the release point will depend on the quantity released, the height of the release point, wind speed, atmospheric stability, heat contained in the release, precipitation on the terrain, physical and chemical form of the released material, and other factors. For a continuous release from an elevated point source under constant meteorological conditions (i.e. wind direction, wind speed and atmospheric stability), the concentration, $C(x, y, z)$, can be calculated from a bivariate Gaussian distribution:

$$C(x,y,z) = \dot{Q} \cdot \frac{\exp(-\frac{y^2}{2\sigma_y(x)^2})}{2\pi\sigma_y(x)\sigma_z(x)u} \cdot \left(\exp\left[-\frac{(z-h)^2}{2\sigma_z(x)^2} \right] + \exp\left[-\frac{(z+h)^2}{2\sigma_z(x)^2} \right] \right) \qquad (1)$$

where

$C(x, y, z)$ is the air concentration at a point (x, y, z),

x is the downwind distance,

y is the crosswind distance,

z is the height above ground,

$\dot{Q}$ is the release rate of the radionuclides,

u is the wind speed,

$\sigma_y(x), \sigma_z(x)$ are the dispersion parameters, which depend on downwind distance and atmospheric stability, and

h is the effective release height.

The above expression for the concentration, C, can be modified to include plume depletion due to dry deposition, wet deposition and radioactive decay [12].

If the influence of the ground on the radiation field is neglected, the dose rate in air at a receptor point (x, y, z) can be calculated by a point kernel integration over the total plume volume:

$$\dot{D}(x,y,z) = E \cdot y \cdot \frac{\mu_{en}}{\rho} \cdot \int_0^\infty \int_{-\infty}^\infty \int_0^\infty \frac{C(x,y,z)}{4\pi r^2} \cdot B(\mu r) \cdot \mathrm{e}^{-\mu r} \mathrm{d}x\mathrm{d}y\mathrm{d}z \qquad (2)$$

where r is the distance from the receptor point to the point source in the infinitesimal plume element $C\mathrm{d}x\mathrm{d}y\mathrm{d}z$, μ is the linear attenuation coefficient for air, B is the dose build-up factor for air, E is the photon energy, y is the photon field and μ_{en}/ρ is the mass-energy-absorption coefficient for air. The numerical volume integration of the Gaussian plume is fairly complex and computer time consuming, and in many cases sufficiently accurate approximations could be made.

Once the radioactive material has reached ground level and is dispersed uniformly throughout a hemisphere several hundred metres in radius with origin at a receptor, the receptor is then assumed to be immersed in the cloud. This assumption, usually referred to as the **semi-infinite cloud** approximation, can be used to calculate the external γ-dose as the product of nuclide specific dose conversion factors (in units of $\mathrm{Sv{\cdot}s^{-1}{\cdot}Bq^{-1}{\cdot}m^3}$) and time-integrated air concentration values (in unit of $\mathrm{Bq{\cdot}s{\cdot}m^{-3}}$) calculated by the atmospheric dispersion and transport model (Eq. 1). The method is an approximation which will give correct results only if certain conditions are fulfilled.

Use of this method can, however, result in an underestimate of the dose up to several kilometres from the release point for a release height of 100 metres and for a stable atmosphere (Pasquill E and F). This underestimate occurs at distances where the material has not yet reached ground level, thereby producing zero or low air concentration values and dose, although the receptor is receiving a dose from the cloudshine. On the other hand, close to the point where the plume first makes contact with the ground, this method can overestimate the dose because the local concentration pattern does not satisfy the semi-infinite cloud assumption of a uniform distribution of material throughout a hemisphere several hundred metres in radius.

If these restrictions are recognized and allowance are made for differences that can occur when dose estimates are made for elevated releases of γ-emitters (see Section 3.1.2), then it is reasonable to use the semi-infinite cloud approximation for dose calculations for external exposure.

Doses from β-emitters can be calculated with little error using the semi-infinite cloud approximation because of the relatively short range of β-particles in air.

3.1.1 β-radiation

β-particles from the decay of radionuclides have a relatively short range in air on the order of a few metres. Therefore, plumes from a release of radionuclides to the atmosphere can be considered as semi-infinite in size when calculating β-doses to the skin from immersion in the plume. The dose rate in air in a given activity concentration, C, can be calculated as:

$$\dot{d}_{\beta,air,\infty} = \frac{\dot{D}_{\beta,air,\infty}}{C} = k \cdot \sum_i \overline{E}_{\beta,i} \cdot p_{\beta,i} = k \cdot \overline{E}_{\beta,total} \tag{3}$$

where

$\overline{E}_{\beta,i}$ is the average β-energy per disintegration for the specific β-decay mode,

$p_{\beta,i}$ is the number of β-particles emitted per disintegration, and

$\overline{E}_{\beta,total}$ is the average β-energy for the different β-decay modes (β-spectra).

The constant, k, has the following value, when it is assumed that the density of air is 1.3 $\mathrm{g{\cdot}cm^{-3}}$:

$$k = 6.2 \cdot 10^{-14} \ \frac{\mathrm{Gy}}{\mathrm{s}} \cdot \frac{\mathrm{m^3}}{\mathrm{Bq}} \cdot \frac{\mathrm{dis}}{\mathrm{MeV}} \tag{4}$$

The β-dose rate to the skin can be calculated from the β-dose rate in air by multiplying the air dose rate with the mass-stopping power ratio for tissue and air:

$$\dot{d}_{\beta,tissue,\infty} = \frac{\dot{D}_{\beta,tissue,\infty}}{C} = k \cdot \sum_i \overline{E}_{\beta,i} \cdot p_{\beta,i} \cdot \left(\frac{(S/\rho)_{tissue}}{(S/\rho)_{air}} \right)_i \quad (5)$$

Most of the body surface is covered by clothing which will attenuate the β-particles and thereby reduce the β-dose rate to the skin. This type of protection is analysed in Section 4.1.3.

The average β-energies,$\overline{E}_{\beta,total}$, for selected radionuclides including those relevant in nuclear reactor accidents are shown in Table 1 [18].

Table 1. Average β-energy, $\overline{E}_{\beta}$, from radioactive decay of β-/γ-emitting radionuclides.

Nuclide	$\overline{E}_{\beta,total}$ [MeV]
^{24}Na	0.554
^{40}K	0.593
^{60}Co	0.0958
^{65}Zn	0.143
^{85}Kr	0.251
^{86}Rb	0.668
^{88}Rb	0.348
^{89}Sr	0.583
^{90}Sr	0.196
^{90}Y	0.934
^{91}Y	0.603
^{95}Nb	0.0434
^{103}Ru	0.0716
^{106}Ru	0.0100
^{106}Rh	1.425
^{132}Te	0.0630
^{131}I	0.183
^{132}I	0.496
^{133}I	0.409
^{135}I	0.375
^{133}Xe	0.100
^{135}Xe	0.303
^{134}Cs	0.157
^{137}Cs	0.178
^{140}La	0.528
^{144}Ce	0.0822
^{144}Pr	1.217

3.1.2 γ-radiation

For γ-radiation the conditions for a semi-infinite cloud geometry are not always fulfilled. If the plume around the receptor point has a reasonably uniform concentration distribution up to a few hundred metres around the point, so-called γ-energy

equilibrium exists, i.e. the photon energy released in a small volume of air is equal to the photon energy absorbed in the same volume. Under such conditions the γ-dose rate in air for a given activity concentration, C, can be calculated from:

$$\dot{d}_{\gamma,air,\infty} = \frac{\dot{D}_{\gamma,air,\infty}}{C} = k \cdot \sum_i E_{\gamma,i} \cdot y_{\gamma,i} \tag{6}$$

where

$E_{\gamma,i}$ is the photon energy per disintegration, and

$y_{\gamma,i}$ is the number of photons emitted per disintegration.

It is here assumed that the influence of the ground on the radiation field can be neglected.

The constant, k, has the following value:

$$k = 6.2 \cdot 10^{-14} \ \frac{\mathrm{Gy}}{\mathrm{s}} \cdot \frac{\mathrm{m}^3}{\mathrm{Bq}} \cdot \frac{\mathrm{dis}}{\mathrm{MeV}} \tag{7}$$

The γ-dose rate in air can also be expressed by the kerma rate constant, Γ, for given radionuclides as:

$$\dot{d}_{\gamma,air,\infty} = \frac{\dot{D}_{\gamma,air,\infty}}{C} \cong 1800 \ \mathrm{m} \cdot \Gamma \tag{8}$$

The dose (kerma) rate in air is expressed in SI-units when Γ and C are expressed in SI-units. The unit of the constant 1800 is expressed in metres.

The gamma dose rate to soft tissue can be calculated from the gamma dose rate in air by multiplying the air dose rate with the mass-energy absorption coefficient ratio for tissue and air:

$$\dot{d}_{\gamma,tissue,\infty} = \frac{\dot{D}_{\gamma,tissue,\infty}}{C} = k \cdot \sum_i E_{\gamma,i} \cdot y_{\gamma,i} \cdot \left(\frac{(\mu_{en}/\rho)_{tissue}}{(\mu_{en}/\rho)_{air}} \right)_i \tag{9}$$

Equation 9 is only valid, if the extension of the tissue is small compared to the mean free path of the photons in tissue. Therefore, the equation can be used only to calculate γ-doses at the surface of the body.

Values of β- and γ-dose rates to soft tissue at the surface of the body from submersion in a semi-infinite cloud per unit nuclide concentration are shown in Table 2 calculated from Equations 5 and 9 [19].

If the conditions for a semi-infinite cloud geometry are unfulfilled it is necessary to correct the calculated dose rate for the semi-infinite geometry. When the concentration at ground level, C, is calculated from the Gaussian plume model, the γ-dose rate in tissue from a finite Gaussian plume can be calculated from [20]:

$$\dot{d}_{\gamma,tissue} = \dot{d}_{\gamma,tissue,\infty} \cdot \exp\left(\frac{1}{2} \cdot \frac{z^2}{\sigma_z^2} \right) \cdot CF(\sigma_z, z/\sigma_z) \tag{10}$$

where z is the height of the plume above ground level and σ_z is the vertical dispersion parameter.

The correction factor $CF(\sigma_z, z/\sigma_z)$ is shown in Fig. 1 as a function of the vertical dispersion parameter, σ_z, for different values of the ratio, z/σ_z [20].

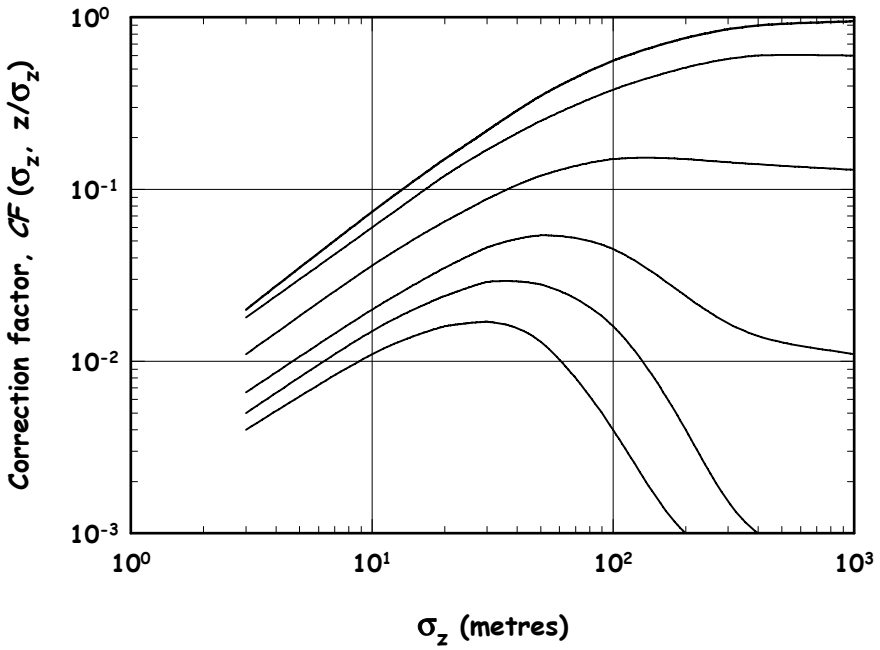

Figure 1. Correction factor, $CF(\sigma_z, z/\sigma_z)$, for the calculation of dose rate at ground level in the centerline of a finite Gaussian plume elevated a distance, z, above ground level. The vertical dispersion parameter, σ_z, is a function of the distance from the release point and the stability category.

If the concentration around the receptor point at ground level is uniformly distributed but the hemisphere is limited in size with radius, R, the correction factor can be expressed as:

$$CF(\mu(E_\gamma) \cdot R) = 1 - \exp(-\mu(E_\gamma) \cdot R) \tag{11}$$

where μ is the linear attenuation coefficient for air. This parameter is a function of the photon energy, E_γ, and, in principle, the correction factor should enter the summation in Eqs. (6) and (9). As an approximation it is normally sufficient to use a single value of μ for the photon energy, namely, one that contributes most to the dose rate.

Table 2. β- and γ-dose rate in tissue at the body surface from submersion in contaminated air of semi-infinite size.

Nuclide	$\dot{d}_{\beta,tissue,\infty}$ [mGy·h^{-1}/MBq·m^{-3}]	$\dot{d}_{\gamma,tissue,\infty}$ [mGy·h^{-1}/MBq·m^{-3}]
^{24}Na	$1.5{\cdot}10^{-1}$	$1.1{\cdot}10^{+0}$
^{60}Co	$2.7{\cdot}10^{-2}$	$6.8{\cdot}10^{-1}$
^{65}Zn	$1.8{\cdot}10^{-3}$	$1.6{\cdot}10^{-1}$
^{85}Kr	$6.9{\cdot}10^{-2}$	$6.0{\cdot}10^{-4}$
^{86}Rb	$1.8{\cdot}10^{-1}$	$2.5{\cdot}10^{-2}$
^{87}Kr	$3.6{\cdot}10^{-1}$	$2.1{\cdot}10^{-1}$
^{88}Rb	$5.6{\cdot}10^{-1}$	$1.8{\cdot}10^{-1}$
^{88}Kr	$9.6{\cdot}10^{-2}$	$5.3{\cdot}10^{-1}$
^{89}Sr	$1.6{\cdot}10^{-1}$	$3.6{\cdot}10^{-5}$
^{90}Sr	$5.4{\cdot}10^{-2}$	0
^{91}Sr	$1.8{\cdot}10^{-1}$	$1.9{\cdot}10^{-1}$
^{90}Y	$2.5{\cdot}10^{-1}$	$4.2{\cdot}10^{-7}$
^{91}Y	$1.7{\cdot}10^{-1}$	$9.7{\cdot}10^{-4}$
^{95}Nb	$1.2{\cdot}10^{-2}$	$2.0{\cdot}10^{-1}$
^{103}Ru	$2.1{\cdot}10^{-2}$	$1.3{\cdot}10^{-1}$
^{106}Ru	$2.8{\cdot}10^{-3}$	0
^{106}Rh	$3.9{\cdot}10^{-1}$	$5.5{\cdot}10^{-2}$
^{132}Te	$2.8{\cdot}10^{-2}$	$6.2{\cdot}10^{-2}$
^{131}I	$5.3{\cdot}10^{-2}$	$1.0{\cdot}10^{-1}$
^{132}I	$1.4{\cdot}10^{-1}$	$6.1{\cdot}10^{-1}$
^{133}I	$1.1{\cdot}10^{-1}$	$1.6{\cdot}10^{-1}$
^{135}I	$1.0{\cdot}10^{-1}$	$4.2{\cdot}10^{-1}$
^{133}Xe	$3.8{\cdot}10^{-2}$	$1.2{\cdot}10^{-2}$
^{135}Xe	$8.7{\cdot}10^{-2}$	$6.6{\cdot}10^{-2}$
^{134}Cs	$4.5{\cdot}10^{-2}$	$4.2{\cdot}10^{-1}$
^{137}Cs	$6.5{\cdot}10^{-2}$	$1.6{\cdot}10^{-1}$
^{140}La	$1.5{\cdot}10^{-1}$	$6.1{\cdot}10^{-1}$
^{144}Ce	$2.5{\cdot}10^{-2}$	$5.4{\cdot}10^{-3}$
^{144}Pr	$3.3{\cdot}10^{-1}$	$8.5{\cdot}10^{-3}$

3.2 Inhalation of radioactive material

A person immersed in the plume would inhale an amount of radioactive material proportional to the time of residence in the plume, the person's breathing rate, and the concentration of radioactive material at his or her location. The inhaled material would then be transported from the lungs to the organs of the body in a manner depending on the nature and chemical form of the radioactive material. The material would eventually be removed, in part by radioactive decay and in part by excretion and other biological processes. The total radiation dose from inhalation would thus be received over a time period that could vary from a few weeks to many years, depending on the chemical and physical nature of the radioactive material and the half-life of the radionuclides.

The dose calculation for inhalation exposure is more straightforward than for external exposure, since the exposure is directly related to the air concentration at the receptor location as a function of time. The time-integrated air concentration (exposure integral) value $\int C \cdot \mathrm{d}t$ ($\mathrm{Bq \cdot s \cdot m^{-3}}$), is directly related to the amount of material available during a time interval Δt for inhalation by an individual at the receptor location with a specified breathing rate, I ($\mathrm{m^3 \cdot s^{-1}}$). Therefore, the total inhaled activity by an individual located at this point is obtained by multiplying the breathing rate by the time-integrated air concentration value. Given the dose conversion factors ($\mathrm{Sv \cdot Bq^{-1}}$) for the particular radionuclides of interest, the inhalation dose can be calculated for the total inhaled activity.

The rate of committed dose, i.e. the committed dose per residence time, from inhalation of a given radionuclide in a radioactive cloud with nuclide concentration, C, can be calculated from:

$$\dot{E}(t_0) = I(t_0) \cdot e(t_0) \cdot C \tag{12}$$

where

$I(t_0)$ is the breathing rate at age t_0, and

$e(t_0)$ is the committed effective dose per unit activity of a given radionuclide inhaled by a person at age t_0.

The breathing rate, I, will depend on age, t_0, and also on the time of day. An adult has an average breathing rate during working hours of 10 $\mathrm{m^3}$/8 h and an average breathing rate during non-working hours of 10 $\mathrm{m^3}$/16 h. The average breathing rate over 24 hours is therefore 20 $\mathrm{m^3}$/24 h corresponding to $2.3 \cdot 10^{-4}$ $\mathrm{m^3 \cdot s^{-1}}$.

Values of $e(t_0)$ for relevant radionuclides in a reactor accident are shown in Table 3 [21]. At present, age-dependent committed effective doses per unit intake are available from the ICRP for only a limited number of radionuclides [22]. A German report, however, contains inhalation and ingestion dose factors for more than 800 radionuclides and 5 age groups. [1]

To describe the clearance of inhaled radioactive materials from the lung, materials are classified with reference to their retention in the pulmonary region [21]. This classification applies to a range of biological half-lives. When $e(t_0)$ for a given radionuclide is given for more than one inhalation class, the highest value of $e(t_0)$ has been used in this report.

It is expected that the ICRP Task Group on the development of a new lung model will present a revised model within a year or two. The inhalation dose factors shown in Table 3 will then be changed accordingly.

[1] K. Henrichs, U. Elsasser, CH. Schotola and A. Kaul, Dosisfaktoren für Inhalation oder Ingestion von Radionuklidverbindungen, ISH 78–81, Institut für Strahlenhygiene, Bundesgesundheitsamt, 8042 Neuhenberg, 1985.

Table 3. Committed effective accumulated dose to age 70 years, $e(t_0)$, from inhalation of airborne radionuclides at different ages, t_0.

Nuclide	3 months	1 year	5 years	15 years	adult
			$e(t_0)$ [mSv/MBq]		
^{24}Na					$2.5{\cdot}10^{-1}$
^{60}Co					$8.3{\cdot}10^{0}$
^{65}Zn					$5.0{\cdot}10^{0}$
^{86}Rb					$1.7{\cdot}10^{0}$
^{88}Rb					$2.5{\cdot}10^{-2}$
^{89}Sr					$1.0{\cdot}10^{1}$
^{90}Sr	$1.2{\cdot}10^{2}$	$1.0{\cdot}10^{2}$	$6.3{\cdot}10^{1}$	$8.9{\cdot}10^{1}$	$6.0{\cdot}10^{1}$
^{91}Sr					$5.0{\cdot}10^{-1}$
^{90}Y					$2.5{\cdot}10^{0}$
^{91}Y					$1.3{\cdot}10^{1}$
^{95}Nb	$1.2{\cdot}10^{1}$	$8.7{\cdot}10^{0}$	$4.9{\cdot}10^{0}$	$2.3{\cdot}10^{0}$	$1.7{\cdot}10^{0}$
^{103}Ru	$2.2{\cdot}10^{1}$	$1.5{\cdot}10^{1}$	$8.0{\cdot}10^{0}$	$3.6{\cdot}10^{0}$	$2.5{\cdot}10^{0}$
^{106}Ru	$9.8{\cdot}10^{2}$	$7.5{\cdot}10^{2}$	$4.2{\cdot}10^{2}$	$1.8{\cdot}10^{2}$	$1.3{\cdot}10^{2}$
^{106}Rh					$5.0{\cdot}10^{-2}$
^{132}Te					$2.5{\cdot}10^{0}$
^{131}I	$6.9{\cdot}10^{1}$	$6.7{\cdot}10^{1}$	$3.9{\cdot}10^{1}$	$1.3{\cdot}10^{1}$	$8.2{\cdot}10^{0}$
^{132}I	$1.0{\cdot}10^{0}$	$8.0{\cdot}10^{-1}$	$4.2{\cdot}10^{-1}$	$1.5{\cdot}10^{-1}$	$9.9{\cdot}10^{-2}$
^{133}I					$1.7{\cdot}10^{0}$
^{135}I					$2.5{\cdot}10^{-1}$
^{134}Cs	$1.7{\cdot}10^{1}$	$9.9{\cdot}10^{0}$	$8.3{\cdot}10^{0}$	$1.3{\cdot}10^{1}$	$1.2{\cdot}10^{1}$
^{137}Cs	$1.3{\cdot}10^{1}$	$7.6{\cdot}10^{0}$	$6.0{\cdot}10^{0}$	$8.7{\cdot}10^{0}$	$8.6{\cdot}10^{0}$
^{140}La					$1.3{\cdot}10^{0}$
^{144}Ce	$7.9{\cdot}10^{2}$	$6.0{\cdot}10^{2}$	$3.3{\cdot}10^{2}$	$1.4{\cdot}10^{2}$	$1.0{\cdot}10^{2}$
^{144}Pr					$1.3{\cdot}10^{-2}$
^{239}Pu	$1.9{\cdot}10^{5}$	$1.7{\cdot}10^{5}$	$1.4{\cdot}10^{5}$	$1.2{\cdot}10^{5}$	$1.2{\cdot}10^{5}$
^{240}Pu					$1.3{\cdot}10^{5}$
^{241}Pu	$2.7{\cdot}10^{3}$	$2.7{\cdot}10^{3}$	$2.5{\cdot}10^{3}$	$2.4{\cdot}10^{3}$	$2.3{\cdot}10^{3}$
^{241}Am	$1.7{\cdot}10^{5}$	$1.5{\cdot}10^{5}$	$1.3{\cdot}10^{5}$	$1.1{\cdot}10^{5}$	$1.1{\cdot}10^{5}$

The committed effective dose accumulated by age 70 years due to inhalation at age, t_0, is calculated as:

$$e(t_0) = \int_{t_0}^{70} \dot{e}(t, t_0)\mathrm{d}t \tag{13}$$

where $\dot{e}(t, t_0)$ is the effective dose rate to an individual at age t and of age t_0 at the time of inhalation.

For long-lived radionuclides with a fast biokinetic distribution and excretion (e.g. ^{137}Cs), the dose would be absorbed within a relatively short period of time. The

reason for the difference between values of $e(t_0)$ in Table 3 for different age groups is that organ masses differ at given ages [22].

The time distribution of $e(t_0)$ would be of interest if the risk of deterministic effects or age-specific stochastic risks are to be evaluated. The methodology for calculating this time distribution is elaborated below.

The effective dose accumulated at the age T from inhalation at age t_0 can be calculated from:

$$e(T, t_0) = \int_{t_0}^{T} \dot{e}(t, t_0) \mathrm{d}t \tag{14}$$

The effective dose rate at age t, $\dot{e}(t, t_0)$, from inhalation at age t_0 can be calculated from the retention function, $R(t)$, as:

$$\dot{e}(t, t_0) = -e(t_0) \cdot \frac{\mathrm{d}R(t)}{\mathrm{d}t} \tag{15}$$

The retention function, $R(t)$, is defined as the fraction of the initial uptake of activity retained in the body at time, t.

As an example, $R(t)$ for cesium can be adequately expressed as a sum of two exponential components of the form [22]:

$$R(t) = [A \cdot \mathrm{e}^{-\lambda_1 \cdot (t - t_0)} + (1 - A) \cdot \mathrm{e}^{-\lambda_2 \cdot (t - t_0)}] \cdot \mathrm{e}^{-\lambda \cdot (t - t_0)} \tag{16}$$

where the parameters A, λ_1 and λ_2 will depend on age t_0. For adults, $A = 0.1$, $\lambda_1 = 0.35$ d^{-1} and $\lambda_2 = 0.0063$ d^{-1}. For the age $t_0 = 5$ years, $A = 0.45$, $\lambda_1 = 0.076$ d^{-1} and $\lambda_2 = 0.023$ d^{-1} [22].

The effective dose accumulated at age T from an intake of ^{137}Cs at age t_0 can thus be calculated from Eqs. (14-16) (neglecting radioactive decay as $\lambda \ll \lambda_2$):

$$e(T, t_0) = e(t_0) \cdot [A \cdot (1 - \mathrm{e}^{-\lambda_1 \cdot (T - t_0)}) + (1 - A) \cdot (1 - \mathrm{e}^{-\lambda_2 \cdot (T - t_0)})] \tag{17}$$

For adults, an intake of ^{137}Cs by inhalation will give the following accumulation in time of the committed effective dose, $e(t_0)$, shown in Table 3:

10% after 4 days

25% after 30 days

50% after 90 days

75% after 200 days

90% after 350 days

For the age group $t_0 = 5$ years the corresponding time distribution of $e(t_0)$ will be:

10% after 2 days

25% after 6 days

50% after 17 days

75% after 38 days

90% after 75 days

3.3 Shielding factors for external γ-radiation

The shielding factor, S, for plume radiation is defined for a given location (indoor or outdoor) as the ratio of dose rate at that location to an outdoor reference dose rate:

$$S = \frac{\dot{D}(x, y, z)}{\dot{D}_{ref}(x, y, z = 1 \text{ m})} \tag{18}$$

Thus, from this definition a low shielding factor afford high protection and vice versa. The reference dose rate, $\dot{D}_{ref}$, is specified at a location 1 meter above ground with the same horizontal position to the plume as that of the location considered, but without any buildings present.

3.3.1 Indoor residence

Shielding factors have been calculated for Nordic single-family houses and multistory buildings [23]. The shielding factors are calculated without the contribution to γ-dose rate from contaminated indoor air. This contribution could increase the value of the shielding factor by 2-8%.

Building types are described as a composite of cubic boxes. The thickness of the different walls and floors/ceiling can be varied independently of each other in the calculation model. The dimensions and building data for Nordic single-family houses and multistory buildings are shown in Table 4 [32].

Table 4. Structure dimensions in kg·m^{-2} for Nordic single-family houses and multistory buildings.

Country	Outer walls	Inner walls	Partition walls	Floors	Roof/ ceiling
	Single-family houses				
Denmark	400	220	-	-	110
Norway	36	20	-	-	81
Sweden	150	50	-	-	100
Finland	50	40	-	-	100
	Multistory buildings				
Denmark	600	220	260	170	250
Norway	390	28	360	450	81
Sweden	300	100	325	425	250
Finland	350	250	450	450	100

A major part of the indoor dose rate from a semi-infinite cloud will originate from the part of the cloud lying above the roof plane if the roof thickness is less than 100 kg·m^{-2} [23].

The calculations of the shielding factors have been based on the structural dimensions shown in Table 4 for a photon energy of 0.7 MeV and for 4 different locations inside the buildings. The shielding factors are shown in Table 5 where the range indicates the dependence of the position inside the building [23].

Table 5. Shielding factors for Nordic housing conditions for γ-radiation from a radioactive plume.

Country	Single-family	Multistory		
		bottom story	3rd story	5th story
Denmark	0.1 (0.05-0.20)	0.03 (0.013-0.056)	0.06 (0.032-0.097)	0.10 (0.081-0.13)
Norway	0.6 (0.56-0.66)	0.05 (0.031-0.069)	0.08 (0.065-0.11)	0.30 (0.27-0.36)
Sweden	0.3 (0.17-0.30)	0.05 (0.026-0.083)	0.08 (0.050-0.12)	0.20 (0.15-0.29)
Finland	0.5 (0.27-0.52)	0.03 (0.016-0.054)	0.05 (0.024-0.12)	0.20 (0.17-0.33)

It appears from Table 5 that Norwegian and Finnish single-family houses offer less protection than Danish and Swedish houses due to their light outer walls. For multistory buildings with 5 stories the best protection is found at the bottom story due to the shielding effect of the upper stories. The poorest protection is found at the top story where the roof has the greatest influence on the shielding factor. Danish houses afford the best protection here because both their roofs and outer walls are relatively heavy.

3.3.2 Outdoor position

At an outdoor position in an urban area the dose rate from a plume would be less than at a position in an open area because the surrounding buildings will act as shielding. The shielding factor will decrease (improving protection) for increasing building height and decreasing street width. Therefore, a densely built-up urban area would provide better outdoor protection than a suburban area. Calculated values of outdoor shielding factors for an urban area are shown in Table 6 [23].

Table 6. Outdoor shielding factors for γ-radiation from a radioactive plume for urban areas with buildings on both sides of the street.

Building height	Street width 10 m	Street width 20 m
10 meter	0.3	0.5
20 meter	0.2	0.3

For an urban area an outdoor shielding factor of 0.5 can be used as a reasonable conservative value. For a suburban area the corresponding value is 0.8.

3.4 Reduction factors for inhalation doses

Depending on the nature of the radioactive materials, a building will provide protection against inhalation of aerosols. Protection of the occupants of a building against inhalation of aerosol-borne activity results from *filtration* in cracks, crevices and pores in the building fabric, by *air exchange* and by further reduction of indoor concentrations through *deposition* upon walls, ceilings, furniture etc. inside the building. However, this latter reduction process has a secondary consequence in

that deposits inside the building may add to the long-term dose commitment from external radiation.

In the simple case of a single compartment and constant air exchange during the passage of an outdoor cloud, the ratio of time integrals of indoor concentration, C_i, to outdoor concentration, C_o is [24]:

$$F = \frac{\int_o^\infty C_i \cdot \mathrm{d}t}{\int_o^\infty C_o \cdot \mathrm{d}t} = f \cdot \frac{\lambda_r}{\lambda_r + \lambda_d} \tag{19}$$

where

f is the infiltration factor (f $\leq$ 1),

λ_d is the deposition rate coefficient, and

λ_r is the air exchange rate coefficient.

The above factor, F, is equal to the instantaneous ratio if the air exchange rate is constant.

The values of the parameters f, λ_d, and λ_r are shown in Table 7 [24] measured for a typical Danish single-family house and for two different particle sizes together with the corresponding reduction factors, F, for indoor inhalation doses.

Table 7. Values of infiltration factor, deposition rate coefficient, air exchange rate constant and inhalation dose reduction factor.

Parameter	Particle size 2 μm	Particle size 4 μm
Infiltration factor, f	0.8 (0.5-1.0)	0.8 (0.5-1.0)
Deposition rate coefficient, λ_d [h^{-1}]	1.0 (0.8-1.2)	2.1 (1.0-2.5)
Air exchange rate coefficient, λ_r [h^{-1}]	0.3 (0.2-1.0)	0.3 (0.2-1.0)
Inhalation dose reduction factor, F	0.2 (0.1-0.5)	0.1 (0.08-0.3)

The above values of the deposition rate coefficient λ_d have been measured in a test house at Risø [25] using an aerosol labelling technique developed at the Imperial College in the UK.

The values of F would increase to about 0.5 if λ_d was as low as 0.2 h^{-1} which corresponds to an average indoor deposition velocity, v_d, of 0.004 cm·s^{-1}. This value of v_d is a factor of 7 below the measured values in the Risø test house. A lower value of the reduction factor, F, can be obtained after a vacuum cleaner is placed in operation inside the house. This effect is equivalent to an increase of the indoor deposition velocity. An extra reduction of the inhalation dose by a factor of 2 could be obtained by this method [24].

In a Finnish study the filtering effect of four flats have been measured, and in a Norwegian study this measurement took place in a wooden single-family house. In both studies the nuclide of ^{7}Be attached to particulate matters was collected on filters simultaneously outside and inside the buildings, and the activity was

determined by γ-spectroscopy. The indoor deposition velocity of ^{7}Be was measured to be 0.04 cm/s for the Norwegian house but no measurements of the deposition velocity were made in the Finnish flats. In the Finnish study the value of F was found to lie in the range 0.23 - 0.45, and the corresponding range in the Norwegian study was 0.40 - 0.47 [36].

4 Exposure from activity deposited on surfaces

4.1 External radiation from deposited activity

Radioactive material can be deposited on the ground by dry or wet deposition and by gravitational settling of particulates. Dry deposition is caused by material in the near surface layer of the plume being deposited on ground level surfaces by impinging on vegetation, buildings and soil. This process continues to deplete the plume as it travels away from the source, and is parameterized in dispersion models by a dry deposition velocity.

Wet deposition is caused by radioactive material coming into contact with a precipitation system (e.g. rain or snow), whereby the radioactive gases and particulates (not the noble gases) are scavenged by the precipitation process and deposited on the ground. This is an efficient scavenging process which can lead to high ground levels of radioactive materials, even at distances far from the release point. Wet deposition is usually parameterized in dispersion models by an exponential depletion of the plume as a function of distance and precipitation rate (wash-out coefficient).

After the material on the ground has been accounted for by each of the deposition processes, the dose at a receptor point is calculated by integrating the dose rate to an individual from each radionuclide deposited on the ground surface. Since most dose conversion factors are given in $\mathrm{Sv{\cdot}s^{-1}{\cdot}Bq^{-1}{\cdot}m^{2}}$, the individual dose rate can be calculated by multiplying the dose conversion factor by the activity level per unit area. Removal processes such as migration into the soil, weathering, radioactive decay etc. should be included in the time integration of the dose rate. The basis for most of the dose conversion factors for ground deposition is the infinite surface assumption, where the receptor point, assumed to be located one meter above the ground surface, is exposed to γ-radiation emitted from radioactive material uniformly distributed on an infinite plane.

4.1.1 Ground surfaces

β-radiation

The β-dose rate in air, $\dot{D}_{\beta,air}$, at a distance, a, above a smooth contaminated ground surface will - due to the relatively short range of β-particles in air - depend only on the surface contamination density within a few meters around the point of detection. Furthermore, the β-dose rate at the distance a is non-zero only if the maximum β-particle range in air is greater than the distance a.

In Table 8 the β-dose rate in air per unit surface contamination density is shown as a function of the average β-energy, $\overline{E}_{\beta}$, and for different values of the distance a between 50 and 170 cm above the ground surface [18]. It appears that the β-dose rate at an average β-energy around 0.3 MeV will decrease by more than a factor of 10 when the height is increased from 50 to 170 cm. With increasing β-energy this variation will be lower.

Clothing will attenuate the β-particles and reduce the dose rate to the skin. This reduction is analysed in Section 4.1.3.

The β-dose rates shown in Table 8 are calculated for a smooth surface with no roughness. For a real surface the roughness will attenuate the β-particles considerably and result in lower β-dose rates than those shown in Table 8. This reduction factor could be a factor of 10 or more.

Table 8. β-dose rate in air, $\dot{d}_{\beta,air}$, from an infinite smooth surface source as a function of average β-energy, $\overline{E}_\beta$.

$\overline{E}_\beta$ (MeV)	$\dot{d}_{\beta,air}$ [mGy·h^{-1}/MBq·m^{-2}]				
	Distance from surface source, a (cm)				
	50 ·	75 ·	100 ·	125 ·	170 ·
0.010	-	-	-	-	-
0.0495	-	-	-	-	-
0.0769	-	-	-	-	-
0.101	$1.8{\cdot}10^{-3}$	$4.0{\cdot}10^{-4}$	$1.0{\cdot}10^{-4}$	-	-
0.307	$4.3{\cdot}10^{-2}$	$2.4{\cdot}10^{-2}$	$1.6{\cdot}10^{-2}$	$8.2{\cdot}10^{-3}$	$3.0{\cdot}10^{-3}$
0.496	$6.6{\cdot}10^{-2}$	$4.5{\cdot}10^{-2}$	$3.2{\cdot}10^{-2}$	$2.3{\cdot}10^{-2}$	$1.2{\cdot}10^{-2}$
0.695	$9.5{\cdot}10^{-2}$	$7.2{\cdot}10^{-2}$	$5.6{\cdot}10^{-2}$	$4.2{\cdot}10^{-2}$	$2.7{\cdot}10^{-2}$
0.824	$1.1{\cdot}10^{-1}$	$8.2{\cdot}10^{-2}$	$6.6{\cdot}10^{-2}$	$5.3{\cdot}10^{-2}$	$3.7{\cdot}10^{-2}$
1.076	$1.2{\cdot}10^{-1}$	$9.4{\cdot}10^{-2}$	$8.0{\cdot}10^{-2}$	$6.7{\cdot}10^{-2}$	$5.0{\cdot}10^{-2}$
1.243	$1.3{\cdot}10^{-1}$	$1.0{\cdot}10^{-1}$	$8.8{\cdot}10^{-2}$	$7.6{\cdot}10^{-2}$	$6.0{\cdot}10^{-2}$
1.425	$1.3{\cdot}10^{-1}$	$1.1{\cdot}10^{-1}$	$9.0{\cdot}10^{-2}$	$8.0{\cdot}10^{-2}$	$6.3{\cdot}10^{-2}$
2.061	$1.4{\cdot}10^{-1}$	$1.2{\cdot}10^{-1}$	$1.1{\cdot}10^{-1}$	$9.5{\cdot}10^{-2}$	$8.1{\cdot}10^{-2}$

γ-radiation

The γ-dose rate in air at a distance, a, above a smooth, uniformly contaminated ground surface of radius, R, with surface contamination density, Q, can be calculated from:

$$\dot{d}_{\gamma,air} = \frac{\dot{D}_{\gamma,air}}{Q} = 2\pi \cdot \ell\mathrm{n}\left(\frac{R}{a}\right) \cdot \Gamma \cong 34 \cdot \Gamma \qquad (20)$$

The value of 34 emerges from a R/a ratio of 200-300. Γ is here the kerma rate constant for the radionuclide under consideration. When Γ and Q are expressed in SI-units the dose rate is in the same units.

In Eq. (20) no consideration is given to the attenuation and build-up of photons in air. At distances within a few hundred metres around the receptor point attenuation and build-up are more or less counterbalanced.

Surface roughness will attenuate the γ-dose rate compared with the dose rate from a smooth surface. Effective γ-dose rates per unit surface contamination density are shown in Table 9 for different radionuclides [26]. The surface roughness is accounted for in the calculations by assuming that the smooth surface is covered with 3 mm of soil [26].

Table 9. Effective dose rate, $\dot{e}_\infty$, from an infinite plane surface source with a surface roughness corresponding to an effective depth of the source in the soil of 3 mm.

Nuclide	$\dot{e}_\infty$ [mSv·h^{-1}/MBq·m^{-2}]
^{24}Na	$8.6 \cdot 10^{-3}$
^{40}K	$3.6 \cdot 10^{-4}$
^{60}Co	$5.8 \cdot 10^{-3}$
^{65}Zn	$1.4 \cdot 10^{-3}$
^{86}Rb	$2.2 \cdot 10^{-4}$
^{88}Rb	$1.4 \cdot 10^{-3}$
^{89}Sr	$2.0 \cdot 10^{-7}$
^{91}Sr	$2.4 \cdot 10^{-3}$
^{92}Sr	$3.6 \cdot 10^{-3}$
^{90}Y	$2.5 \cdot 10^{-10}$
^{91}Y	$8.3 \cdot 10^{-6}$
^{92}Y	$5.8 \cdot 10^{-4}$
^{95}Nb	$1.8 \cdot 10^{-3}$
^{103}Ru	$1.1 \cdot 10^{-3}$
^{106}Ru/^{106}Rh	$5.0 \cdot 10^{-4}$
^{132}Te	$5.8 \cdot 10^{-3}$
^{131}I	$9.4 \cdot 10^{-4}$
^{132}I	$5.4 \cdot 10^{-3}$
^{133}I	$1.5 \cdot 10^{-3}$
^{135}I	$4.3 \cdot 10^{-3}$
^{134}Cs	$3.6 \cdot 10^{-3}$
^{137}Cs	$1.4 \cdot 10^{-3}$
^{140}La	$5.4 \cdot 10^{-3}$
^{144}Ce	$1.2 \cdot 10^{-4}$
^{144}Pr	$7.2 \cdot 10^{-5}$

4.1.2 Other surfaces

After deposition in an urban environment different radionuclide concentration will be found on the various surfaces of buildings and their surroundings. For dry deposition the relative distribution will depend primarily on the deposition velocities of the radionuclides for the various kinds of surfaces. Vegetation areas like lawns will be more densely contaminated than smooth areas like streets and walls. In particular, a heavy contamination would be expected for trees in full foliage. For wet deposition the relative contamination density will primarily depend on the exposure of the surfaces to the rain and on the run-off of the contamination with rainwater.

Knowledge of the relative contamination of the various urban surfaces is necessary in order to assess dose rates from the deposited activity on these surfaces. Furthermore, as the contamination on the surfaces will decrease with time due to weathering processes, this time dependence should be known for the calculation of doses integrated over time.

Calculations of dose rates have been made [27] for single-family houses and house blocks in compact urban surroundings. The results are shown in Tables 10 and 11.

Table 10. γ-dose rates at locations inside and outside a single-family house from deposited activity on various surfaces.

Deposition area, i	Gamma dose rate, $\dot{d}_{ij}$ [mGy·h^{-1}/MBq·m^{-2}] 0.3 MeV	0.662 MeV	3 MeV
	Basement		
Windows	$7.2{\cdot}10^{-8}$	$5.4{\cdot}10^{-7}$	$8.6{\cdot}10^{-6}$
Walls	$5.4{\cdot}10^{-8}$	$5.4{\cdot}10^{-7}$	$1.9{\cdot}10^{-5}$
Roof	$4.0{\cdot}10^{-7}$	$2.6{\cdot}10^{-6}$	$3.8{\cdot}10^{-5}$
Ground	$3.6{\cdot}10^{-8}$	$2.9{\cdot}10^{-7}$	$4.0{\cdot}10^{-6}$
Neighboring buildings	$1.1{\cdot}10^{-8}$	$7.2{\cdot}10^{-8}$	$2.2{\cdot}10^{-6}$
Trees	$1.4{\cdot}10^{-8}$	$1.3{\cdot}10^{-7}$	$3.6{\cdot}10^{-6}$
	Ground floor		
Windows	$2.4{\cdot}10^{-5}$	$5.4{\cdot}10^{-5}$	$1.8{\cdot}10^{-4}$
Walls	$2.2{\cdot}10^{-5}$	$7.2{\cdot}10^{-5}$	$5.0{\cdot}10^{-4}$
Roof	$3.2{\cdot}10^{-5}$	$7.6{\cdot}10^{-5}$	$3.1{\cdot}10^{-4}$
Ground	$6.6{\cdot}10^{-5}$	$1.7{\cdot}10^{-4}$	$1.0{\cdot}10^{-3}$
Neighboring buildings	$1.3{\cdot}10^{-5}$	$3.6{\cdot}10^{-5}$	$1.8{\cdot}10^{-4}$
Trees	$1.4{\cdot}10^{-5}$	$4.0{\cdot}10^{-5}$	$1.9{\cdot}10^{-4}$
	Outside		
Windows	$3.2{\cdot}10^{-6}$	$7.2{\cdot}10^{-6}$	$2.9{\cdot}10^{-5}$
Walls	$2.2{\cdot}10^{-4}$	$4.5{\cdot}10^{-4}$	$1.2{\cdot}10^{-3}$
Roof	$2.9{\cdot}10^{-6}$	$8.3{\cdot}10^{-6}$	$7.9{\cdot}10^{-5}$
Ground	$7.6{\cdot}10^{-4}$	$1.6{\cdot}10^{-3}$	$4.5{\cdot}10^{-3}$
Neighboring buildings	$2.3{\cdot}10^{-4}$	$4.3{\cdot}10^{-4}$	$1.1{\cdot}10^{-3}$
Trees	$1.1{\cdot}10^{-4}$	$2.2{\cdot}10^{-4}$	$6.8{\cdot}10^{-4}$

Table 11. γ-dose rates at locations inside and outside a multistory house block with four floors from deposited activity on various surfaces.

Deposition area, i	Gamma dose rate, $\dot{d}_{ij}$ [mGy·h^{-1}/MBq·m^{-2}]		
	0.3 MeV	0.662 MeV	3 MeV
	Ground floor		
Windows	$1.0{\cdot}10^{-5}$	$2.3{\cdot}10^{-5}$	$8.6{\cdot}10^{-5}$
Walls	$1.8{\cdot}10^{-6}$	$7.6{\cdot}10^{-6}$	$9.4{\cdot}10^{-5}$
Roof	-	-	$3.6{\cdot}10^{-8}$
Street	$3.6{\cdot}10^{-6}$	$9.4{\cdot}10^{-6}$	$8.6{\cdot}10^{-5}$
Neighboring buildings	$7.2{\cdot}10^{-6}$	$2.0{\cdot}10^{-5}$	$1.2{\cdot}10^{-4}$
	Fourth floor		
Windows	$1.0{\cdot}10^{-5}$	$2.3{\cdot}10^{-5}$	$8.6{\cdot}10^{-5}$
Walls	$1.4{\cdot}10^{-6}$	$7.2{\cdot}10^{-6}$	$9.4{\cdot}10^{-5}$
Roof	$2.2{\cdot}10^{-6}$	$1.4{\cdot}10^{-5}$	$2.0{\cdot}10^{-4}$
Street	$3.2{\cdot}10^{-7}$	$5.4{\cdot}10^{-7}$	$4.0{\cdot}10^{-6}$
Neighboring buildings	$5.0{\cdot}10^{-6}$	$1.3{\cdot}10^{-5}$	$8.7{\cdot}10^{-5}$
	Outside street		
Windows	$2.9{\cdot}10^{-5}$	$5.8{\cdot}10^{-5}$	$1.7{\cdot}10^{-4}$
Walls	$2.1{\cdot}10^{-4}$	$4.1{\cdot}10^{-4}$	$1.1{\cdot}10^{-3}$
Roof	$7.2{\cdot}10^{-7}$	$1.1{\cdot}10^{-6}$	$2.5{\cdot}10^{-6}$
Street	$7.2{\cdot}10^{-4}$	$1.5{\cdot}10^{-3}$	$4.5{\cdot}10^{-3}$
Neighboring buildings	$4.8{\cdot}10^{-4}$	$9.8{\cdot}10^{-4}$	$2.9{\cdot}10^{-3}$

4.1.3 Body surface

If β-/γ-emitting radionuclides are deposited uniformly on the body surface, the major part of the dose rate to the skin will originate from β-particles. The β-particle range in soft tissue is less than 2 cm for most β-energies that occur in radioactive decay. Therefore, a uniformly contaminated area on the body surface as small as 15 cm^2 is regarded as effectively infinite in estimating the β-dose to skin at depths below the center of the surface contamination.

Calculated values of the β-dose rate to the skin at a depth of 7 mg·cm^{-2} are shown in Table 12 for a number of β-/γ-emitting nuclides [28]. The basal layer of the skin is covered by a dead skin layer of varying thickness. An average value of 7 mg·cm^{-2} for the whole body surface is used here.

Table 12. β-dose rate to skin, $\dot{d}_{\beta,skin}$, at a depth of seven milligram per cm^2 from uniformly deposited activity on the skin surface.

Nuclide	$\dot{d}_{\beta,skin}$ [mGy·h^{-1}/MBq·m^{-2}]
^{24}Na	$2.3 \cdot 10^{-1}$
^{60}Co	$1.1 \cdot 10^{-1}$
^{65}Zn	$2.6 \cdot 10^{-3}$
^{86}Rb	$2.3 \cdot 10^{-1}$
^{88}Rb	$2.6 \cdot 10^{-1}$
^{89}Sr	$2.3 \cdot 10^{-1}$
^{90}Sr	$1.8 \cdot 10^{-1}$
^{91}Sr	$2.3 \cdot 10^{-1}$
^{90}Y	$2.4 \cdot 10^{-1}$
^{91}Y	$2.3 \cdot 10^{-1}$
^{95}Nb	$2.6 \cdot 10^{-2}$
^{103}Ru	$6.6 \cdot 10^{-2}$
^{106}Ru/^{106}Rh	$2.5 \cdot 10^{-1}$
^{132}Te	$8.0 \cdot 10^{-2}$
^{131}I	$1.7 \cdot 10^{-1}$
^{132}I	$2.2 \cdot 10^{-1}$
^{133}I	$2.2 \cdot 10^{-1}$
^{135}I	$2.1 \cdot 10^{-1}$
^{134}Cs	$1.4 \cdot 10^{-1}$
^{137}Cs	$1.6 \cdot 10^{-1}$
^{140}Ba	$1.9 \cdot 10^{-1}$
^{140}La	$2.3 \cdot 10^{-1}$
^{144}Ce	$1.0 \cdot 10^{-1}$
^{144}Pr	$2.5 \cdot 10^{-1}$

Most of the body is covered by clothing. The total body surface of an ICRP standard man is 1.8 m^2. The mass of light summer clothing is around 0.4 kg and the mass of winter clothing is up to around 2.5 kg (Nordic countries). Averaging these masses over the total body surface gives an average mass-thickness range of 0.02-0.14 g·cm^{-2} for summer-winter clothing.

For β-radiation the mass-attenuation coefficient, μ_m, for low-Z materials can be expressed by the maximum β-energy, $E_{\beta,max}$. This expression is valid within a β-energy range of 0.1-4 MeV:

$$\mu_m(E_{\beta,max}) \cong 17 \text{ cm}^2 \cdot \text{g}^{-1} \cdot \text{MeV}^{-1} \cdot E_{\beta,max}^{-1.14} \tag{21}$$

The protection factor, PF, for the β-dose as a function of mass-thickness and β-energy can then be calculated from:

$$PF(E_{\beta,max}, x_m) = \text{e}^{-\mu_m(E_{\beta,max}) \cdot x_m} \tag{22}$$

The protection factor is shown in Fig. 2 as a function of the maximum β-energy and for 3 different values of the mass-thickness, x_m.

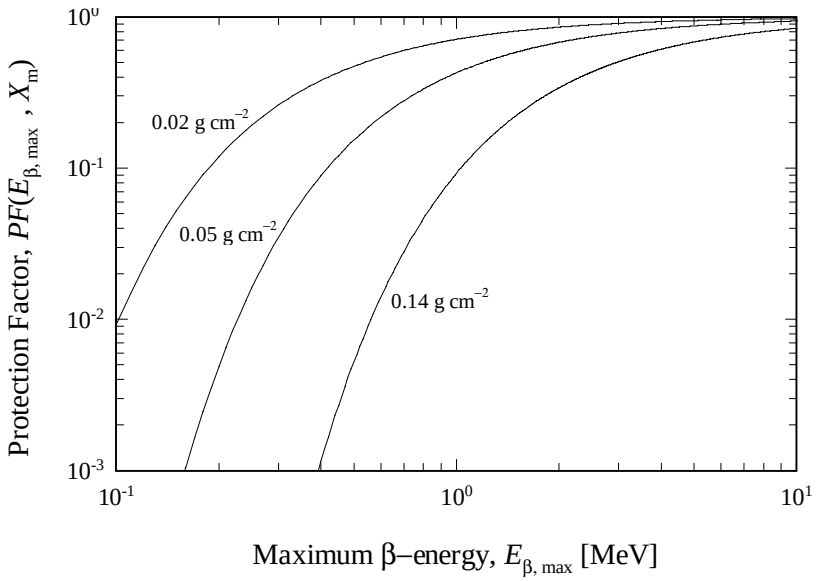

Figure 2. Protection factor for β-radiation as a function of β-energy for 3 different values of the mass-thickness, x_m.

Most of the radionuclides in a reactor accident will have maximum β-energies on the order of 0.3-0.5 MeV. Representative values of the protection factor, PF, will - according to Fig. 2 - be for summer, spring/autumn and winter clothing approximately 0.2-0.3, 0.1 and 0.001, respectively.

4.2 Location factors for external γ-radiation

Location factors are defined as the ratio of the dose rate at the respective location inside or outside the buildings and a reference dose rate. This factor is often called a shielding factor or reduction factor. However, since the deposition density in the environment, especially with trees present, may be higher than for the reference location, this term is not entirely adequate. The factor which accounts for the modification of the reference dose rate at different locations is therefore called the *location factor* [29], defined as:

$$L_j = \frac{\sum_i s_i \cdot \dot{d}_{ij}}{\dot{D}_{ref}} \tag{23}$$

where $\dot{d}_{ij}$ is the dose rate at location j from deposition area i and s_i is the relative surface contamination density of area i.

Knowledge of the relative contamination density of the various urban surfaces after a deposition of radionuclides is relatively uncertain, in particular for windows, roofs and trees. However, some experimental experience was gained from the Chernobyl accident and estimated values of s_i are shown in Table 13 for dry and wet deposition [30].

Table 13. Estimate of relative surface contamination densities, s_i, of different urban surfaces shortly after a deposition in summer relative to a fresh dry deposition on a lawn.

Deposition area	Dry deposition	Wet deposition
Windows	0.03	0.01
Walls	0.1	0.02
Roof	0.3	0.7
Lawn	1.0	0.8
Street	0.1	0.4
Trees	5.0	0.1
Indoor surfaces	0.01	0.005

Based on the γ-dose rates, $\dot{d}_i$, per unit surface contamination density shown in Tables 10 and 11 and the relative surface contamination densities, s_i, shown in Table 13, location factors can be calculated from Eq. (23).

4.2.1 Location factors for urban areas

Typical location factors to γ-radiation from dry deposited activity have been calculated for Nordic housing conditions [31, 32, 35] and shown in Table 14. In these calculations only ground surfaces, streets, walls and roofs have been considered as deposition areas. For suburban areas the surface contamination densities on walls and roofs have been assumed to be 10% of that on the ground (lawn). For multistory buildings in urban environments, equal surface contamination densities on walls, roofs and streets have been assumed. This is in accordance with the values shown in Table 13 except for roofs which have a higher density than streets and walls.

Table 14. Location factors for Nordic housing conditions to γ-radiation from activity deposited on ground and building surfaces.

Country	Single-family	Multistory		
		bottom story	3rd story	5th story
Denmark	0.04-0.30	0.018-0.089	0.006-0.032	0.029-0.13
Norway	0.31	0.08	0.02	0.11
Sweden	0.23	0.04	0.01	0.03
Finland	0.36	0.03	0.01	0.04

The location factors shown in Table 14 have been calculated for a photon energy of 0.5 MeV and with the building data for Nordic single-family houses and multistory buildings shown in Table 4 [32]. Parameter variations have been made for the Danish buildings [35] as shown in Table 14.

For the multistory buildings there is a significant difference between the best shielded locations (the middle stories) and the poorest (bottom or top story, depending on the roof thickness).

A comparison of the location factors in Table 14 with the shielding factors for plume radiation in Table 5 indicates that for single-family houses the shielding factors for plume γ-radiation are approximately a factor of 2 larger than the corresponding location factors for γ-radiation from deposited activity. The reason is that shielding factors for plume radiation depend mainly on the roof thickness while the location factors for radiation from deposited activity depend mainly on the outer wall thickness, which normally is larger than that of the roof.

4.2.2 Location factors for vehicles

Location factors for cars and busses to γ-radiation from activity deposited on buildings and ground surfaces have been determined experimentally [33]. The dose rate reduction inside different vehicles with and without passengers was measured. The contaminated ground and building surfaces were simulated by a relatively strong point source (150 GBq) of ^{137}Cs that was placed in a matrix around the vehicle out to a distance of 100 metres and up to a height of 10 metres.

In the calculations of the measurement data it was assumed that the surface contamination densities in urban areas were the same on all surfaces. In the experiments, the street width was simulated to be 15 metres and the building height at both side of the street 10 metres. The calculated location factors from the measurements are shown in Table 15 [33].

Table 15. Location factors for cars and busses moving in urban areas to γ-radiation from activity deposited on streets and building surfaces.

Vehicle type	Without passengers	With passengers
Cars	0.35 (0.30-0.40)	0.30 (0.25-0.35)
Busses	0.25 (0.20-0.30)	0.20 (0.15-0.30)

The values of the location factors are lower if passengers are present in the vehicles. The reason is the mutual shielding effect of the body.

5 Importance of the different exposure pathways

Radiation doses to man originating from the different plume exposure pathways will depend strongly on the radionuclides in the release. Doses from a release of actinides such as plutonium will be dominated by the inhalation and ingestion pathways because the external radiation from actinides normally is insignificant and because the α-radiation will give large internal doses per unit intake.

Radiation doses from a release of β-/γ-emitting radionuclides will originate from both internal (inhalation and ingestion) and external exposure. The relative contribution from each exposure pathway will again depend on the radionuclides.

The *direct* exposure pathways for β-/γ-emitting radionuclides released to the atmosphere will be:

(a) External γ-doses to the whole body from radioactive material in the plume.

(b) External β-/γ-doses to the skin from radioactive material in the plume.

(c) Internal doses from inhalation of radioactive material in the plume.

(d) External γ-doses to the whole body from radioactive material deposited on surfaces of the ground, buildings etc.

(e) External β-/γ-doses to the skin from radioactive material deposited on surfaces of the ground, buildings etc.

(f) External β-doses to the skin from radioactive material deposited on the body surface.

The *indirect* exposure pathways such as ingestion of contaminated foodstuffs and inhalation of resuspended material are dealt with in the project BER-121 [34].

To reveal the relative importance of the different exposure pathways, calculations of doses from a release of the different radionuclides shown in the preceding tables have been made. The assumptions for these calculations are that the release of each of the radionuclides considered will cause a concentration, C, in the air at ground level of 1 MBq·m^{-3}. The release time, T, and dry-deposition velocity, v_d, have to be specified to perform the calculations. The surface contamination density at the receptor point is thus $C \cdot v_d \cdot T$ just after the plume passage.

5.1 External γ-dose to the whole body from the plume passage

The external γ-dose to the body surface is calculated from:

$$D_{\gamma,tissue} = C \cdot T \cdot \dot{d}_{\gamma,tissue,\infty} \cdot S \cdot \exp\left(\frac{1}{2} \cdot \frac{z^2}{\sigma_z^2}\right) \cdot CF(\sigma_z, z/\sigma_z) \tag{24}$$

If the cloud at the receptor point can be considered as semi-infinite the calculation will be simplified to:

$$D_{\gamma,tissue} = C \cdot T \cdot \dot{d}_{\gamma,tissue,\infty} \cdot S \tag{25}$$

The effective dose/surface dose ratio will be around 0.7 for external γ-radiation and the effective dose from the plume passage will therefore be:

$$E_{plume} = 0.7 \cdot C \cdot T \cdot \dot{d}_{\gamma,tissue,\infty} \cdot S \tag{26}$$

5.2 External β-/γ-dose to skin from the plume passage

The external β-/γ-dose to the uncovered skin of a person staying at an outdoor location during the plume passage is calculated as the sum of the β- and γ-doses to the skin:

$$D_{\beta+\gamma,skin} = C \cdot T \cdot (\dot{d}_{\gamma,tissue,\infty} \cdot S + \dot{d}_{\beta,tissue,\infty}) \tag{27}$$

If the plume is not semi-infinite in size the skin dose will be reduced to:

$$D_{\beta+\gamma,skin} = C \cdot T \cdot \left(\dot{d}_{\gamma,tissue,\infty} \cdot S \cdot \exp\left(\frac{1}{2} \cdot \frac{z^2}{\sigma_z^2}\right) \cdot CF(\sigma_z, z/\sigma_z) + \dot{d}_{\beta,tissue,\infty}\right) \tag{28}$$

At an indoor residence the β-dose will be zero from the outdoor plume due to the shielding effect of building materials, and a factor of F (Table 7) lower than the outdoor dose due to a lower indoor concentration.

5.3 Inhalation dose from the plume passage

The committed effective dose to a person of age t_0 from inhalation of radioactive material during the plume passage at an outdoor location is calculated as:

$$E(t_0) = C \cdot T \cdot I \cdot e(t_0) \tag{29}$$

At an indoor residence the committed effective dose will be:

$$E(t_0) = C \cdot T \cdot I \cdot e(t_0) \cdot F \tag{30}$$

5.4 External γ-dose rate to the whole body from radioactive material deposited on surfaces

The effective dose rate just after the plume passage at an outdoor position from radioactive material deposited on a infinite ground surface without buildings within a distance of several hundred meters around the receptor point will be:

$$\dot{E} = C \cdot v_d \cdot T \cdot \dot{e}_\infty \tag{31}$$

At a given residence, j, (indoor or outdoor) in an urban environment the effective dose rate from each of the different surfaces i with the relative surface contamination density s_i will be:

$$\dot{E}_{ij} = C \cdot v_d \cdot T \cdot s_i \cdot \dot{d}_{ij} \tag{32}$$

5.5 External β-/γ-dose rate to skin from radioactive material deposited on ground surfaces

The external β-/γ-dose rate to the uncovered skin of a person staying at an outdoor position is calculated as the sum of the β- and γ-dose rates one meter above ground level. The β-dose rate in air from deposited activity (Table 8) can be used as a first approximation of the skin β-dose rate. Similarly, the effective dose rate (Table 9) from deposited activity can be used as an approximation of the skin γ-dose rate:

$$\dot{D}_{\beta+\gamma,skin} \cong C \cdot v_d \cdot T \cdot (\dot{e}_\infty + \dot{d}_{\beta,air}) \tag{33}$$

At an indoor residence the β-dose rate will be zero (at zero indoor surface concentration) due to the shielding effect of the building materials. The indoor γ-dose rate to the skin can be calculated from:

$$\dot{D}_{\gamma,skin} \simeq \dot{E}_{ij} \tag{34}$$

where $\dot{E}_{ij}$ is calculated from Eq. (32) from indoor positions.

5.6 External β-dose rate to skin from radioactive material deposited on the body surface

The external β-dose rate to the skin from radioactive material deposited on the uncovered skin can be calculated from:

$$\dot{D}_{\beta,skin} = C \cdot v_d \cdot T \cdot \dot{d}_{\beta,skin} \tag{35}$$

For the body surface covered by clothing the β-dose rate will be:

$$\dot{D}_{\beta,skin} = C \cdot v_d \cdot T \cdot \dot{d}_{\beta,skin} \cdot PF \tag{36}$$

It is assumed here that the surface contamination density, $C \cdot v_d \cdot T$, is the same for the body and ground surfaces.

5.7 Comparison of doses from the different exposure pathways

The doses and dose rates from the different exposure pathways have been calculated under the assumption that the concentration, C, of the different radionuclides in the air at ground level during the plume passage is 1 MBq·m^{-3}. The following parameter values have been used in the calculations:

Release time, T: 2 hours

Dry-deposition velocity, v_d: 0.001 m·s^{-1}

Plume size: semi-infinite

Outdoor shielding factor for plume radiation, S: 0.5

Indoor shielding factor for plume radiation, S: 0.1

Indoor location factor for ground radiation, L: 0.07

Age at time of inhalation, t_0: adult ($t_0 > 18$ years)

Reduction factor for inhalation dose indoor, F: 0.3

Breathing rate, I: $2.3 \cdot 10^{-4}$ m^3·s^{-1}

Tables 16 and 17 list the calculated doses and dose rates for an outdoor residence and Tables 18 and 19 for an indoor residence.

Table 16. Effective doses and dose rates at an outdoor position from a 2-hour release of different radionuclides to the atmosphere. The outdoor concentration, C, at the receptor point is 1 MBq·m^{-3} during the plume passage, and the dry-deposition velocity, v_d, is 0.001 m·s^{-1}.

Effective doses and dose rates from different plume exposure pathways

Nuclide	Plume pathways E_{plume} [mSv]	Plume pathways $E(t_0)$ [mSv]	Surface pathways $\dot{E}$ [mSv·h^{-1}]
^{60}Co	$4.8{\cdot}10^{-1}$	$1.4{\cdot}10^{+1}$	$4.2{\cdot}10^{-2}$
^{85}Kr	$4.2{\cdot}10^{-4}$	-	-
^{87}Kr	$1.5{\cdot}10^{-1}$	-	-
^{88}Kr	$3.7{\cdot}10^{-1}$	-	-
^{89}Sr	$2.5{\cdot}10^{-5}$	$1.7{\cdot}10^{+1}$	$1.4{\cdot}10^{-6}$
^{90}Sr	-	$9.9{\cdot}10^{+1}$	-
^{91}Sr	$1.3{\cdot}10^{-1}$	$8.3{\cdot}10^{-1}$	$1.7{\cdot}10^{-2}$
^{90}Y	-	$4.1{\cdot}10^{+0}$	-
^{91}Y	$6.8{\cdot}10^{-4}$	$2.2{\cdot}10^{+1}$	$6.0{\cdot}10^{-5}$
^{95}Nb	$1.4{\cdot}10^{-1}$	$2.8{\cdot}10^{+0}$	$1.3{\cdot}10^{-2}$
^{103}Ru	$9.1{\cdot}10^{-2}$	$4.1{\cdot}10^{+0}$	$7.9{\cdot}10^{-3}$
^{106}Ru	-	$2.2{\cdot}10^{+2}$	-
^{131}I	$7.0{\cdot}10^{-2}$	$1.4{\cdot}10^{+1}$	$6.8{\cdot}10^{-3}$
^{132}I	$4.3{\cdot}10^{-1}$	$1.6{\cdot}10^{-1}$	$3.9{\cdot}10^{-2}$
^{133}I	$1.1{\cdot}10^{-1}$	$2.8{\cdot}10^{+0}$	$1.1{\cdot}10^{-2}$
^{135}I	$2.9{\cdot}10^{-1}$	$4.1{\cdot}10^{-1}$	$3.1{\cdot}10^{-2}$
^{133}Xe	$8.4{\cdot}10^{-3}$	-	-
^{135}Xe	$4.6{\cdot}10^{-2}$	-	-
^{134}Cs	$2.9{\cdot}10^{-1}$	$2.0{\cdot}10^{+1}$	$2.6{\cdot}10^{-2}$
^{137}Cs	$1.1{\cdot}10^{-1}$	$1.4{\cdot}10^{+1}$	$1.0{\cdot}10^{-2}$
^{144}Ce	$3.8{\cdot}10^{-3}$	$1.7{\cdot}10^{+2}$	$8.6{\cdot}10^{-4}$
^{144}Pr	$6.0{\cdot}10^{-3}$	$2.2{\cdot}10^{-2}$	$5.2{\cdot}10^{-4}$
^{239}Pu	-	$2.0{\cdot}10^{+5}$	-

Table 17. Skin doses and dose rates at an outdoor position from a 2-hour release of different radionuclides to the atmosphere. The outdoor concentration, C, at the receptor point is 1 MBq·m^{-3} during the plume passage, and the dry-deposition velocity, v_d, is 0.001 m·s^{-1}.

Skin doses and dose rates from different plume exposure pathways

Nuclide	Plume pathways	Surface pathways	
	$D_{\beta+\gamma,skin}$ [mSv]	$\dot{D}_{\beta+\gamma,skin}$ [mSv·h^{-1}]	$\dot{D}_{\beta,skin}$ [mSv·h^{-1}]
^{60}Co	$7.3{\cdot}10^{-1}$	$4.2{\cdot}10^{-2}$	$7.9{\cdot}10^{-1}$
^{85}Kr	-	-	
^{87}Kr	$9.3{\cdot}10^{-1}$	-	-
^{88}Kr	$7.2{\cdot}10^{-1}$	-	-
^{89}Sr	$3.2{\cdot}10^{-1}$	$2.9{\cdot}10^{-1}$	$1.7{\cdot}10^{+0}$
^{90}Sr	$1.1{\cdot}10^{-1}$	$7.2{\cdot}10^{-3}$	$1.3{\cdot}10^{+0}$
^{91}Sr	$5.5{\cdot}10^{-1}$	$1.3{\cdot}10^{-1}$	$1.7{\cdot}10^{+0}$
^{90}Y	$5.0{\cdot}10^{-1}$	$5.0{\cdot}10^{-1}$	$1.7{\cdot}10^{+0}$
^{91}Y	$3.4{\cdot}10^{-1}$	$2.9{\cdot}10^{-1}$	$1.7{\cdot}10^{+0}$
^{95}Nb	$2.2{\cdot}10^{-1}$	$1.3{\cdot}10^{-2}$	$1.9{\cdot}10^{-1}$
^{103}Ru	$1.7{\cdot}10^{-1}$	$7.9{\cdot}10^{-3}$	$4.8{\cdot}10^{-1}$
^{106}Ru	$5.6{\cdot}10^{-3}$	-	-
^{131}I	$2.1{\cdot}10^{-1}$	$1.0{\cdot}10^{-2}$	$1.2{\cdot}10^{+0}$
^{132}I	$8.9{\cdot}10^{-1}$	$2.7{\cdot}10^{-1}$	$1.6{\cdot}10^{+0}$
^{133}I	$3.8{\cdot}10^{-1}$	$1.5{\cdot}10^{-1}$	$1.6{\cdot}10^{+0}$
^{135}I	$6.2{\cdot}10^{-1}$	$1.7{\cdot}10^{-1}$	$1.5{\cdot}10^{+0}$
^{133}Xe	$8.8{\cdot}10^{-2}$	-	-
^{135}Xe	$2.4{\cdot}10^{-1}$	-	-
^{134}Cs	$5.1{\cdot}10^{-1}$	$3.3{\cdot}10^{-2}$	$1.0{\cdot}10^{+0}$
^{137}Cs	$2.9{\cdot}10^{-1}$	$1.7{\cdot}10^{-2}$	$1.2{\cdot}10^{+0}$
^{144}Ce	$5.5{\cdot}10^{-2}$	$8.6{\cdot}10^{-4}$	$7.2{\cdot}10^{-1}$
^{144}Pr	$6.7{\cdot}10^{-1}$	$6.3{\cdot}10^{-1}$	$1.8{\cdot}10^{+0}$
^{239}Pu	-	-	-

Table 18. Effective doses and dose rates at an indoor residence from a 2-hour release of different radionuclides to the atmosphere. The outdoor concentration, C, at the receptor point is 1 MBq·m^{-3} during the plume passage, and the dry deposition-velocity, v_d, is 0.001 m·s^{-1}.

Effective doses and dose rates from different plume exposure pathways

Nuclide	Plume pathways		Surface pathways
	E_{plume}	$E(t_0)$	$\dot{E}$
	[mSv]		[mSv·h^{-1}]
^{60}Co	$9.5{\cdot}10^{-2}$	$4.1{\cdot}10^{+0}$	$2.9{\cdot}10^{-3}$
^{85}Kr	$8.4{\cdot}10^{-5}$	-	-
^{87}Kr	$2.9{\cdot}10^{-2}$	-	-
^{88}Kr	$7.4{\cdot}10^{-2}$	-	-
^{89}Sr	$5.0{\cdot}10^{-6}$	$5.0{\cdot}10^{+0}$	$1.0{\cdot}10^{-7}$
^{90}Sr	-	$3.0{\cdot}10^{+1}$	-
^{91}Sr	$2.7{\cdot}10^{-2}$	$2.5{\cdot}10^{-1}$	$1.2{\cdot}10^{-3}$
^{90}Y	-	$1.2{\cdot}10^{+0}$	-
^{91}Y	$1.4{\cdot}10^{-4}$	$6.5{\cdot}10^{+0}$	$4.2{\cdot}10^{-6}$
^{95}Nb	$2.8{\cdot}10^{-2}$	$8.4{\cdot}10^{-1}$	$9.1{\cdot}10^{-4}$
^{103}Ru	$1.8{\cdot}10^{-2}$	$1.2{\cdot}10^{+0}$	$5.5{\cdot}10^{-4}$
^{106}Ru	-	$6.5{\cdot}10^{+1}$	-
^{131}I	$1.4{\cdot}10^{-2}$	$4.1{\cdot}10^{+0}$	$4.7{\cdot}10^{-4}$
^{132}I	$8.5{\cdot}10^{-2}$	$4.9{\cdot}10^{-2}$	$2.7{\cdot}10^{-3}$
^{133}I	$2.2{\cdot}10^{-2}$	$8.4{\cdot}10^{-1}$	$7.6{\cdot}10^{-4}$
^{135}I	$5.9{\cdot}10^{-2}$	$1.2{\cdot}10^{-1}$	$2.2{\cdot}10^{-3}$
^{133}Xe	$1.7{\cdot}10^{-3}$	-	-
^{135}Xe	$9.2{\cdot}10^{-3}$	-	-
^{134}Cs	$5.9{\cdot}10^{-2}$	$6.0{\cdot}10^{+0}$	$1.8{\cdot}10^{-3}$
^{137}Cs	$2.2{\cdot}10^{-2}$	$4.3{\cdot}10^{+0}$	$7.1{\cdot}10^{-4}$
^{144}Ce	$7.6{\cdot}10^{-4}$	$5.0{\cdot}10^{+1}$	$6.0{\cdot}10^{-5}$
^{144}Pr	$1.2{\cdot}10^{-3}$	$6.5{\cdot}10^{-3}$	$3.6{\cdot}10^{-5}$
^{239}Pu	-	$6.0{\cdot}10^{+4}$	-

Table 19. Skin doses and dose rates at an indoor residence from a 2-hour release of different radionuclides to the atmosphere. The outdoor concentration, C, at the receptor point is 1 MBq·m^{-3} during the plume passage, and the dry deposition-velocity, v_d, is 0.001 m·s^{-1}.

Skin doses and dose rates from different plume exposure pathways

Nuclide	Plume pathways $D_{\beta+\gamma,skin}$ [mSv]	Surface pathways $\dot{D}_{\beta+\gamma,skin}$ [mSv·h^{-1}]	$\dot{D}_{\beta,skin}$ [mSv·h^{-1}]
^{60}Co	$1.4\cdot10^{-1}$	$2.9\cdot10^{-3}$	$2.4\cdot10^{-1}$
^{85}Kr	$1.2\cdot10^{-4}$	-	-
^{87}Kr	$4.2\cdot10^{-2}$	-	-
^{88}Kr	$1.1\cdot10^{-1}$	-	-
^{89}Sr	$7.2\cdot10^{-6}$	$1.0\cdot10^{-7}$	$5.0\cdot10^{-1}$
^{90}Sr	-	-	$3.9\cdot10^{-1}$
^{91}Sr	$3.8\cdot10^{-2}$	$1.2\cdot10^{-3}$	$5.0\cdot10^{-1}$
^{90}Y	-	-	$5.2\cdot10^{-1}$
^{91}Y	$1.9\cdot10^{-4}$	$4.2\cdot10^{-6}$	$5.0\cdot10^{-1}$
^{95}Nb	$4.0\cdot10^{-2}$	$9.1\cdot10^{-4}$	$5.6\cdot10^{-1}$
^{103}Ru	$2.6\cdot10^{-2}$	$5.5\cdot10^{-4}$	$1.4\cdot10^{-1}$
^{106}Ru	-	-	-
^{131}I	$2.0\cdot10^{-2}$	$4.7\cdot10^{-4}$	$3.7\cdot10^{-1}$
^{132}I	$1.2\cdot10^{-1}$	$2.7\cdot10^{-3}$	$4.8\cdot10^{-1}$
^{133}I	$3.2\cdot10^{-2}$	$7.6\cdot10^{-4}$	$4.8\cdot10^{-1}$
^{135}I	$8.4\cdot10^{-2}$	$2.2\cdot10^{-3}$	$4.5\cdot10^{-1}$
^{133}Xe	$2.4\cdot10^{-3}$	-	-
^{135}Xe	$1.3\cdot10^{-2}$	-	-
^{134}Cs	$8.4\cdot10^{-2}$	$1.8\cdot10^{-3}$	$3.0\cdot10^{-1}$
^{137}Cs	$3.2\cdot10^{-2}$	$7.1\cdot10^{-4}$	$3.5\cdot10^{-1}$
^{144}Ce	$1.1\cdot10^{-3}$	$6.0\cdot10^{-5}$	$2.2\cdot10^{-1}$
^{144}Pr	$1.7\cdot10^{-3}$	$3.6\cdot10^{-5}$	$5.4\cdot10^{-1}$
^{239}Pu	-	-	-

From the results of the dose calculations some information on the relative importance of the different exposure pathways can be extracted. It should be emphasized, however, that this information is of indicative nature only and that more precise analyses should be made for specific accidents.

Effective γ-doses from plume and ground

External γ-doses will originate from both the plume passage (cloudshine) and from radioactive material deposited on ground surfaces (groundshine). The ratio of the effective dose from cloudshine, E_{plume}, for a given nuclide to the initial dose rate from groundshine, $\dot{E}$, for the same nuclide has been calculated to be:

$$\frac{E_{plume}}{\dot{E}} \approx \begin{cases} 10 \text{ h for an outdoor location} \\ 30 \text{ h for an indoor residence} \end{cases} \tag{37}$$

The implication of this ratio is that 10 - 30 hours after the plume has passed, the effective dose accumulated from groundshine would be equal to the dose already received from cloudshine.

Effective doses from external γ-radiation and from inhalation
Inhalation doses from plume exposure will be absorbed over a period that could vary from a few weeks to many years after the plume has passed (committed dose). The ratio of the committed effective dose, $E(t_0)$, for a given nuclide to the effective dose from cloudshine, E_{plume}, from the same nuclide has been calculated to be:

$$\frac{E(t_0)}{E_{plume}} \approx 10 - 100 \tag{38}$$

for both indoor and outdoor positions. Exceptions are the nuclides ^{89}Sr, ^{91}Y and ^{144}Ce for which the ratio is much higher.

External β-/γ-dose rates to the skin from deposited activity
The external dose rate to the skin will originate from radionuclides deposited on the body and ground surfaces. The skin dose rate from a specific radionuclide deposited on the body surface will be dominated completely by β-radiation, whereas the skin dose rate from groundshine will be due to both β- and γ-radiation. The ratio of the β-dose rate to skin, $\dot{D}_{\beta,skin}$, from a body-deposited radionuclide to the β-/γ-dose rate to skin, $\dot{D}_{\beta+\gamma,skin}$, from groundshine from the same nuclide has been calculated to be:

$$\frac{\dot{D}_{\beta,skin}}{\dot{D}_{\beta+\gamma,skin}} \approx \begin{cases} \text{5 - 50 for an outdoor location} \\ \text{50 - 200 for an indoor residence} \end{cases} \tag{39}$$

At an indoor position there will be no contribution from β-groundshine because all the β-particles will be absorbed in the building materials.

Committed effective external and inhalation doses
The committed effective inhalation dose, $E(t_0)$, from the plume will depend on the biokinetic of the inhaled material. The committed effective external dose from material deposited on the ground surface will depend on the environmental removal rate of the deposited material. The ratio of the committed effective external dose to the committed effective inhalation dose for the radionuclides ^{60}Co, ^{134}Cs and ^{137}Cs has been calculated to be:

$$\frac{\int_0^{70} \dot{E} \cdot \mathrm{d}t}{E(t_0)} \approx \begin{cases} \text{30 - 100 for an outdoor location} \\ \text{10 - 30 for an indoor residence} \end{cases} \tag{40}$$

In the above calculations the environmental removal half-life has been assumed to be 10 years.

The calculated ratio for nuclides with half-lives of hours to months would be on the order of 0.1 - 3 for an outdoor location and 0.02 - 0.6 for an indoor one.

Initial γ-dose rate from ^{137}Cs deposited on suburban surfaces

The external γ-dose rates from *dry-deposited* ^{137}Cs on different suburban surfaces have been calculated for the values of the relative surface contamination densities, s_i, shown in Table 13 and the γ-dose rates from deposited activity on various surfaces, d_i, shown in Table 10. The results are given in Table 20.

Table 20. Initial outdoor and indoor effective dose rates, $\dot{E}_i$, in a suburban area from dry deposited ^{137}Cs on different surfaces from a 2-hour release of ^{137}Cs to the atmosphere. The outdoor concentration, C, at the receptor point is 1 MBq·m^{-3} during the plume passage, and the dry-deposition velocity to the ground surface, v_d, is 0.001 m·s^{-1}.

Dose rate, $\dot{E}_i$ [mSv·h^{-1}], from dry-deposited ^{137}Cs in a suburban environment		
Deposition area, i	Outdoor	Indoor
Windows	$1.3{\cdot}10^{-6}$ (0.01 %)	$9.9{\cdot}10^{-6}$ (0.4 %)
Walls	$2.8{\cdot}10^{-4}$ (1.6 %)	$4.4{\cdot}10^{-5}$ (1.8 %)
Roof	$1.5{\cdot}10^{-5}$ (0.09 %)	$1.4{\cdot}10^{-4}$ (5.8 %)
Ground	$9.8{\cdot}10^{-3}$ (57.5 %)	$1.0{\cdot}10^{-3}$ (41.4 %)
Neighboring buildings	$2.6{\cdot}10^{-4}$ (1.5 %)	$2.2{\cdot}10^{-5}$ (0.9 %)
Trees	$6.7{\cdot}10^{-3}$ (39.3 %)	$1.2{\cdot}10^{-3}$ (49.7 %)
Total	$1.7{\cdot}10^{-2}$ (100 %)	$2.4{\cdot}10^{-3}$ (100 %)

At an outdoor position in a suburban area approximately 60% of the initial dose rate originates from the ground area. If trees are present in the area, up to 40% of the initial dose rate could come from deposited cesium on the trees.

At an indoor residence the contribution to the initial indoor dose rate from surrounding trees could be up to 50%. Approximately 40% would come from the deposited cesium on the ground surface and the remaining 10% from deposited cesium on roof, walls and windows.

The relatively high contribution to both outdoor and indoor dose rate from deposited material on trees is due to a high dry-deposition velocity because of a high surface-to-volume ratio.

The γ-dose rates from *wet-deposited* ^{137}Cs on different suburban surfaces have also been calculated, and the results are shown in Table 21.

Table 21. Initial outdoor and indoor effective dose rates, $\dot{E}_i$, in a suburban area from wet-deposited ^{137}Cs on different suburban surfaces. The initial surface contamination density on the ground from wet deposition is assumed to be equal to the initial surface contamination density on the ground from dry deposition.

Dose rate, $\dot{E}_i$ [mSv·h^{-1}], from wet-deposited ^{137}Cs in a suburban environment		
Deposition area, i	Outdoor	Indoor
Windows	$4.4{\cdot}10^{-7}$ (0.01 %)	$3.3{\cdot}10^{-6}$ (0.3 %)
Walls	$5.5{\cdot}10^{-5}$ (0.7 %)	$8.8{\cdot}10^{-6}$ (0.7 %)
Roof	$3.6{\cdot}10^{-5}$ (0.4 %)	$3.3{\cdot}10^{-4}$ (27.5 %)
Ground	$7.8{\cdot}10^{-3}$ (96.3 %)	$8.3{\cdot}10^{-4}$ (69.2 %)
Neighboring buildings	$5.3{\cdot}10^{-5}$ (0.7 %)	$4.4{\cdot}10^{-6}$ (0.4 %)
Trees	$1.3{\cdot}10^{-4}$ (1.6 %)	$2.4{\cdot}10^{-5}$ (2.0 %)
Total	$8.1{\cdot}10^{-3}$ (100 %)	$1.2{\cdot}10^{-3}$ (100 %)

The assumptions made for the calculations of dose rates from wet-deposited material are that the initial surface contamination density from wet deposition on the ground is equal to the initial surface contamination density from dry deposition on the ground. However, for lawns the relative surface contamination density for wet deposition is around 20% less than for dry deposition because of penetration of the radionuclides into the soil with rainwater [29]. The dose rate from the ground is therefore 20% less in case of wet-deposited activity as can be seen in Table 20 and 21.

The distribution of the dose rates from wet-deposited activity on the different suburban surfaces is clearly different from that for dry-deposited activity. It appears from Table 21 that the outdoor dose rate originates almost exclusively from material deposited on the ground. The indoor dose rate is dominated by material deposited on ground and roof surfaces.

The absolute value of the wet-deposited contamination density on the ground surface might be significantly higher than that from dry deposition. This was clearly demonstrated after the Chernobyl accident where the "hot spot" areas in the affected republics were caused by wet deposition. The dose rates from wet-deposited material could therefore be significantly higher than the values in Table 21, which are based on an initial ground surface contamination density from wet deposition which is equal to the initial surface contamination density from dry deposition.

In the Tables 20 and 21 the external exposure pathway from dry deposition on indoor surfaces has been neglected. However, with a relative indoor deposition of 0.03 compared to a lawn, the indoor deposited activity would contribute with about two third of the external dose at a location having a location factor of 0.05.

6 Summary and conclusions

A compilation of data for calculating doses from six direct exposure pathways following an atmospheric release of radioactive material has been made. Many of these data originate from the latest research programmes within the CEC and the Nordic countries as well as from the experience gained after the Chernobyl accident.

The methodologies for calculating doses for emergency planning purposes have been evaluated. The methods can easily be used in an emergency situation as all the exposure pathways are related to the concentration, C, of different radionuclides in air. This concentration could result either from measurements or from calculations based on plume or puff models.

The **committed effective inhalation doses** from the plume passage will dominate over the external cloudshine dose. For most β-/γ-emitting radionuclides the committed inhalation doses would be one to two orders of magnitude higher than the external cloudshine γ-dose. The nuclides ^{90}Sr/^{90}Y, ^{106}Ru/^{106}Rh and ^{144}Ce/^{144}Pr are of special importance as the resulting committed effective inhalation doses are roughly one order of magnitude higher than from other β-/γ-emitting nuclides for equal exposure integrals. The committed effective inhalation doses from the actinides are even higher. For ^{239}Pu, the committed effective inhalation dose would be around three orders of magnitude higher than those from ^{90}Sr/^{90}Y, ^{106}Ru/^{106}Rh and ^{144}Ce/^{144}Pr for equal exposure integrals.

The **external γ-doses** are dominated by groundshine. After the plume has passed, the whole body dose from most of the nuclides accumulated over roughly one day from groundshine from dry deposited activity would be equal to the whole body dose already received from the cloudshine. For wet deposited activity, the groundshine dose might equal the cloudshine dose after only a few hours.

The **committed effective external doses** from groundshine for γ-emitting radionuclides with a long half-life (years) would be at least one order of magnitude higher than the committed effective inhalation dose from the same nuclides. For radionuclides with a half-life of the order of hours to months the committed effective external dose from groundshine would typically be less than the committed effective inhalation dose.

The initial effective dose rate from dry- and wet-deposited nuclides on different suburban areas are dominated by ground, tree and roof surfaces. For **dry-deposited** ^{137}Cs, the outdoor γ-dose rate hierarchy would be **ground/trees/walls** and **trees/ground/roof** for the indoor γ-dose rate.

For **wet-deposited** ^{137}Cs on different suburban surfaces the corresponding outdoor γ-dose rate hierarchy would be **ground/trees/walls** and for the indoor γ-dose rate **ground/roof/trees**.

A reduction of the deposited ^{137}Cs on ground surfaces from decontamination by a factor of 10, which probably is impossible for a large-scale decontamination campaign, would reduce the time-averaged indoor/outdoor γ-dose rate from dry- and wet-deposited ^{137}Cs by roughly a factor of 2 - 4. A more realistic reduction by a factor of 2 of ground-deposited ^{137}Cs by decontamination would reduce the time-averaged dose rate by roughly a factor of 1.5.

The **external skin dose rates** from radionuclides deposited on the ground would be one to two orders of magnitude lower than those originating from the same nuclides deposited directly on the body surface with the same contamination density. The body surface covered by clothing would be exposed to a much lower dose rate than the uncovered skin because of the shielding effect of the clothing for β-radiation. For summer clothing, this reduction factor would be of the order of 3 - 5 and for winter clothing around 1000.

7 References

1. Radiation Protection Programme. Final Report (1985-1989). EUR-13268, 1991.
2. Environmental Consequences of Releases from Nuclear Accidents. A Nordic Perspective. Final report of the NKA Project AKTU-200. Nordic Liaison Committee for Atomic Energy. March 1990.
3. Theoretical Possibilities and Consequences of Major Accidents in Large Nuclear Power Plants. WASH-740, US Atomic Energy Commission, March 1957.
4. Närforlägning av kärnkraftverk. Betänkande av Närförläggningsutredningen. SOU 1974:56, Stockholm 1974 (In Swedish).
5. Reactor Safety Study. An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants. WASH-1400, US Nuclear Regulatory Commission, October 1975.
6. Deutsche Risikostudie Kernkraftwerke - Eine Untersuchung zu dem durch Störfalle in Kernkraftwerken verursachten Risiko. Bundesminister für Forschung und Technologie (Hsg.). Verlag TÜV-Rheinland (1979).
7. Kjernekraft og Sikkerhet. Norges Offentlige Utredninger. NOU 1978:35 A/B), Oslo 1978 (In Norwegian).
8. Miettinen Jaakko, Savolainen Ilkka, Silvennoinen Pekka, Tornio Eija and Seppo Vuori, Risk-benefit evaluation of nuclear power plant siting. Annals of Nuclear Energy, Vol. 3 1976 (pp. 489-500).
9. Ove Edlund and Christina Gyllander, HS-77 Havaristudie - Barsebäck. Konsekvensberäkning. Studsvik/SM-78/5 (In Swedish).
10. Per Hedemann Jensen, Erik Lundtang Petersen, Søren Thykier-Nielsen and Frits Heikel Vinther, Calculation of the Individual and Population Doses on Danish Territory resulting from Hypothetical Core-melt Accidents at the Barsebäck Reactor. Risø-R-356, October 1977.
11. Henry L. Gjørup, Niels Otto Jensen, Per Hedemann Jensen, Ole John Nielsen, Leif Kristensen, Erik Lundtang Petersen, Torben Petersen, Jørn Roed, Søren Thykier-Nielsen, Frits Heikel Vinther and Asker Aarkrog, Radioactive Contamination of Danish Territory after Core-melt Accidents at the Barsebäck Power Plant. Risø-R-462, March 1982.
12. Søren Thykier-Nielsen, Modeller til beregning af gammadoser og inhalationsdoser fra frigørelse af radioaktive stoffer. Risø-M-1725, July 1974 (In Danish).
13. Per Hedemann Jensen, Sammenligning af matematiske modeller til beregning af eksterne gammadoser hidrørende fra radioaktivitetsfrigørelser til atmosfæren. Risø-M-1726, July 1974 (In Danish).
14. Seppo Vuori, Evaluation of nuclear power plant siting by probabilistic assessment of environmental impacts. Technical Research Centre of Finland. Electrical and Nuclear Technology, Publication 24, 1978, Espoo.
15. Ilkka Savolainen, Probabilistic siting analysis of nuclear power plants emphasizing atmospheric dispersion of radioactive releases and radiation-induced health effects. Helsinki University of Technology, 1980.

16. International Comparison Study on Reactor Accident Consequence Modeling. Summary Report to CSNI by an NEA Group of experts. OECD/NEA, September 1984.

17. The International Chernobyl Project. Assessment of Radiological Consequences and Evaluation of Protective Measures. Technical Report. Part G. Report by an International Advisory Committee. IAEA, 1991.

18. W.G. Cross, H. Ing, N.O. Freedman and J. Mainvill, Tables of beta-ray dose distributions in water, air and other media. Chalk River Nuclear Laboratories. AECL-7617, September 1982.

19. D.C. Kocher, Dose-rate conversion factors for external exposure to photon and electron radiation from radionuclides occurring in routine releases from nuclear fuel cycle facilities. Health Physics Vol. 38, April 1980 (pp. 543-621).

20. Reactor Safety Study. An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants. Appendix VI, WASH-1400, US Nuclear Regulatory Commission, October 1975.

21. International Commission on Radiological Protection, Limits for Intakes of Radionuclides by Workers, ICRP Publication 30, July 1978.

22. International Commission on Radiological Protection, Age-dependent Doses to Members of the Public from Intake of radionuclides: Part I, ICRP Publication 56, April 1989.

23. Henry L. Gjørup, Per Hedemann Jensen and Søren Thykier-Nielsen, Shielding Factor Calculation for Plume Radiation. Final Report. I: Radiation Protection Programme. Progress Report 1985-89. Volume 3. EUR-13268, 1991 (pp. 3061-3072).

24. J. Roed, A.J.H. Goddard, C. Lange, J. MacCurtain and M. Byrne, Reduction of Dose from Radioactive Matter Ingressed in Buildings. International seminar on Intervention Levels and Countermeasures for Nuclear Accidents. Cadarache, 7-11 October 1991.

25. J. Roed and A.J.H. Goddard, Ingress of Radioactive Material into Dwellings. Seminar on Methods and Codes for Assessing the Off-site Consequences of Nuclear Accidents. Athens, 7-11 May 1990 (pp. 433-450).

26. P. Jacob, H.G. Paretzke, H. Rosenbaum and Maria Zankl, Organ Doses from Radionuclides on the Ground. Part I: Simple Time Dependences. Health Physics Vol. 54, No. 6, June 1988 (pp. 617-633).

27 . R. Meckbach, P. Jacob and H.G. Paretzke, Gamma Exposures due to Radionuclides Deposited in Urban Environments. Part I: Kerma Rates from Contaminated Urban Surfaces. Radiation Protection Dosimetry Vol. 25, No. 3, 1988 (pp. 167-179).

28. D.C. Kocher and K.F. Eckerman, Electron Dose-rate Conversion factors for External Exposure of the Skin from Uniformly Deposited Activity on the Body Surface. Health Physics Vol. 53, No. 2, August 1987 (pp. 135-141).

29. R. Meckbach and P. Jacob, Gamma Exposures due to Radionuclides Deposited in Urban Environments. Part II: Location Factors for Different Deposition Patterns. Radiation Protection Dosimetry Vol. 25, No. 3, 1988 (pp. 181-190).

30. P. Jacob and R. Meckbach, External Exposure from Deposited Radionuclides. Seminar on Methods and Codes for Assessing the Off-site Consequences of Nuclear Accidents. Athens, 7-11 May 1990 (pp. 407-422).

31. Per Hedemann Jensen, Shielding Factors for Gamma Radiation from Activity Deposited on Structures and Ground Surfaces. Risø-M-2270, November 1982.

32. Ulf Tveten, Shielding factors for Typical Houses in the Nordic Countries. IFE/KR/E-82/001 V, September 1982.

33. Bente Lauridsen and Per Hedemann Jensen, Shielding Factors for Vehicles to Gamma Radiation from Activity Deposited on Structures and Ground Surfaces. Risø-M-2339, April 1982.

34. Ulla Bergström, Atmospheric Dispersion and Environmental Consequences. Exposure from Ground Deposition Pathways. NKS Project BER-121 (In Progress).

35. Per Hedemann Jensen, Calculated Shielding Factors for Selected European Houses. Risø-M-2474, December 1984.

36. G.C. Christensen and R. Mustonen, The Filtering Effect of Buildings on Airborne Particulate Matter. IFE/KR/E-87/002, June 1987.

37. Søren Thykier-Nielsen, Comparison of Nordic Dose Models. Risø-M-1972, April 1978.

Bibliographic Data Sheet **Risø-M-2849(EN)**

Title and author(s)

Atmospheric Dispersion and Environmental Consequences Exposure from Radioactive Plume Pathways

Per Hedemann Jensen

ISBN

87-550-1613-8

ISSN

0418-6453

Dept. or group

Section of Applied Health Physics, Department of Safety

Date

November 1992

Journal No.	Project/contract No.
02696	BER-12(92)1

Pages	Tables	Illustrations	References
42	21	2	37

Abstract (Max. 2000 char.)

Methods are described for assessing early radiation doses due to atmospheric releases of radionuclides, i.e. inhalation and external exposure from the plume and from deposited activity. Data to be used in these assessments are presented. The purpose of the present work is to evaluate methods and data that could be used in emergency situations as well as for emergency planning purposes.

The most important direct pathways following a release of airborne radionuclides to the atmosphere are the inhalation pathway and the external exposure pathway from ground-deposited activity. For long-lived radionuclides like ^{134}Cs and ^{137}Cs the committed effective external dose from deposited activity is 1-2 orders of magnitude larger than the committed effective dose from inhalation. Similarly, the committed effective dose from inhalation is 1-2 orders of magnitude larger than the external γ-dose originating directly from the plume.

Descriptors INIS/EDB

CALCULATION METHODS; DOSE RATES; ENVIRONMENTAL EXPOSURE PATHWAYS; EXTERNAL IRRADIATION; FALLOUT DEPOSITS; INHALATION; RADIOACTIVE CLOUDS; RADIOISOTOPES

Available on request from:

Risø Library
Risø National Laboratory
P.O.Box 49, DK-4000 Roskilde, Denmark
Phone +45 42 37 12 12, ext. 2268/2269
Telex 43116, Telefax +45 46 75 56 29

ISBN 87-550-1613-8
ISSN 0418-6453